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GENERAL

13564 KAPL-101

Knolls Atomic Power Lab., Schenectady, N. Y.

PROGRESS REPORT NO. 26 [FOR] SEPTEMBER 1–30, 1948. Decl. Apr. 4, 1957. 97p. Contract W-31-109-Eng-52. $15 30(ph OTS); $5.40(mf OTS).

Physics. Reactor theory studies are presented on: Doppler effect; multi-group calculation for the Preliminary Pile Assembly (PPA-3); Pu production distribution in the blanket; adjoint calculation and control effect calculation for PPA-3; leakage through local voids in thick shields; and radiation protection associated with unloading machine parts in Schenectady Reactor. Nuclear Measurements. Studies are presented on the Pu alpha experiment and activation experiment. Chemistry. Studies on Chemistry of reactor components include the determination of O_2 in sodium by the amalgam and AgCl method. Separations Chemistry-Redox Process. The following investigations are given: sulfate complexing of U; effect of $Al(NO_3)_3$ on sulfate complexing; extraction of Th by thenoyltrifluoracetone (TTA); sulfate complexing of Th; Pu oxidation kinetics; fission product chemistry of Ru and Ce; conductivity of process solutions; fluorimetric determination of U; fission product analysis of Ru; chelation of Hf with TTA; and Pu valence states. Research Problems. Progress is reported on the following: interaction of metals with Na; solubility of Fe in liquid Na and related thermodynamic and kinetic studies; crystal structure of graphite and β-U: U filled Mo pins; U alloy studies: heat treatment of U and orientation of U; and Ni plating of Be. Metallurgical Engineering. Studies are reported on the following: leak, radiographic, and ultrasonic testing; erosion of Be in Na; static corrosion; inert arc welding; pin sealing; brazing of stainless steel; resistance welding; and creep testing. Data are included on Xe transients in intermediate reactors and thermal stress analysis. (W.L.H.)

BIOLOGY AND MEDICINE

RADIATION EFFECTS

13565 UCLA-195(Del.)

California. Univ., Los Angeles. Atomic Energy Project.

QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1952. Apr. 1, 1952. Decl. with deletions Mar. 7, 1957. 71p. Contract AT-04-1-GEN-12. $12.30(ph OTS); $4.50(mf OTS).

Progress is reported in medical research studies. (C.H.)

13566 UCLA-357

California. Univ., Los Angeles. Atomic Energy Project.

QUARTERLY PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1955. Jan. 1, 1956. Decl. Mar. 19, 1957. 115p. Contract AT-04-1-GEN-12. $18.30(ph OTS); $6.00 (mf OTS).

Progress is reported in medical research studies. (C.H.)

RADIATION HAZARDS AND PROTECTION

13567 AMES-1

New York Operations Office, AEC.

OCCUPATIONAL EXPOSURE TO THORIUM AND BERYLLIUM [AT] AMES RESEARCH LABORATORY. Paul B. Klevin. July 14, 1952. Decl. Mar. 12, 1957. 23p. $4.80 (ph OTS); $2.70(mf OTS).

This preliminary survey presents data on the health and safety problems existing during refining and production of Th metal, from which an estimation of the daily average exposure can be determined for personnel. A brief study was also made of several Be operations. Dust-sampling procedures for Th, thoron, and Be, and analytical procedures for radiation measurements are given. From the data obtained, recommendations are presented for the reduction of the airborne contaminations from Th and Be. (C.H.)

13568 CF-56-10-88

Oak Ridge National Lab., Tenn.

PERSONNEL RADIATION EXPOSURES IN THE THOREX PILOT PLANT. PAPER PRESENTED AT CHEMICAL TECHNOLOGY DIVISION INFORMATION MEETING, OCTOBER 11-12, 1956. G. S. Sadowski. Oct. 19, 1956. Decl. Mar. 13, 1957. 8p. Contract [W-7406-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Radiation dosages received by operating personnel during 21 months of Thorex Pilot Plant operation are tabulated. Lower exposures may be obtained by extensive use of remote-operation equipment and additional shielding. (C.W.H.)

13569 HW-23093

Hanford Works, Richland, Wash.

ACCUMULATION OF RADIOACTIVITY IN COLUMBIA RIVER FISH IN THE VICINITY OF THE HANFORD WORKS. P. A. Olson, Jr. and R. F. Foster. July 1, 1952. Decl. Feb. 19, 1957. 59p. Contract W-31-109-Eng-52. $9.30(ph OTS); $3.60(mf OTS).

13570 MLM-272

Mound Lab., Miamisburg, Ohio.

EFFECTIVE HALF-LIFE OF POLONIUM IN THE HUMAN. (Information Report). David H. Naimark. Jan. 4, 1949. Decl. Apr. 5, 1957. 15p. Contract AT-33-1-gen-53. $3.30(ph OTS); $2.40(mf OTS).

A method of determining the effective half-life of Po in the human is described. In a study of the urinary excretion data of eighteen human cases, calculations based on the analyses of the urine samples submitted result in an overall average effective half-life of 31.4 days. Because of the many influencing factors affecting this study, an approximate thirty-day effective human half-life is suggested, until such time as controlled experiments and more efficient procedures can permit a more accurate determination. (auth)

13571 Y-836

Carbide and Carbon Chemicals Co. Y-12 Plant, Oak Ridge, Tenn.

HEALTH PHYSICS PROGRESS REPORT [FOR] JANUARY 1, 1951–JUNE 30, 1951. E. G. Struxness. Dec. 15, 1951. Decl. Mar. 7, 1957. 38p. Contract W-7405-eng-26. $6.30 (ph OTS); $3.00(mf OTS).

Control methods and decontamination procedures used in the rupture of a Po–Be neutron source container are discussed. Shielding data for the 86 in. cyclotron are given. Plastic rings to be changed weekly were designed to be used for monitoring β radiation in the presence of weak γ radiation. Contents lists and photographs of catastrophe kits are presented. Modifications of an air sampler for installation in a discharge pipe for monitoring effluent air are described. (F.S.)

TRACER APPLICATIONS

13572 HW-41519
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
EVALUATION OF BIOLOGICAL HAZARDS FROM RUTHENIUM PARTICULATES. I. STUDIES OF PERCUTANEOUS ABSORPTION, GASTROINTESTINAL ABSORPTION, AND GASTROINTESTINAL HOLDUP. R. C. Thompson, O. L. Hollis, and W. D. Oakley. Mar. 1, 1956. Decl. Mar. 29, 1957. 25p. Contract W-31-109-Eng-52. $0.35(OTS).

Studies are described of the percutaneous absorption and gastrointestinal absorption in the rat of Ru administered in the form of insoluble particulates. Results are also reported on the gastrointestinal holdup of these particulates. All results are characterized by extreme variability, due, no doubt, to variation in the chemical and physical nature of the particles employed. On the average, absorption of Ru in this form is 100-fold less than the absorption of Ru administered in solution. There was no tendency for prolonged holdup of particles in the gastrointestinal tract. (auth)

CHEMISTRY

13573 A-2354
[Standard Oil Co. of Indiana, Chicago.]
INVESTIGATION OF THE SYSTEM METHYL ETHER AND METHYL BORATE–BORON FLUORIDE COMPLEX. D. A. McCaulay and W. L. Rittschof. Aug. 2, 1945. Decl. Mar. 12, 1957. 21p. Contract W-7418-eng-41. $4.80(ph OTS); $2.70(mf OTS).

It was observed that the separation of methyl borate–boron fluoride complex from dimethyl ether was much more difficult than would be expected from the vapor pressures of the two components. Vapor pressures of the system methyl borate complex-methyl ether were measured over a wide composition range and at various temperatures. (auth)

13574 AGC-AE-19(Pt. I)
Aerojet General Corp., Azusa, Calif.
THE REACTION OF MOLTEN METAL WITH WATER. Progress Report [for] May 1 through June 30, 1956. July 23, 1956. Decl. Jan. 15, 1957. 13p. Contract AT(04-3)-44. (AGC-L2914-17(Pt.I)). $3.30(ph OTS); $2.40(mf OTS).

Experiments were initiated to determine the effect of sudden mixing of molten metal with water. Molten-metal drop tests and corrosion tests were carried out on Zr, Zircaloy-2, and Zircaloy-B with 0.9% Be added. Oxide film thicknesses were determined and photomicrographs were made. Results are presented of molten-metal spray tests which were carried out on Al and Zr. (W.L.H.)

13575 ANL-4215(Del.)
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-I, SUMMARY REPORT FOR JULY, AUGUST, AND SEPTEMBER 1948. Oct. 18, 1948. Decl. with deletions Feb. 1957. 16p. Contract W-31-109-eng-38. $3.30(ph OTS); $2.40(mf OTS).

Study of Ru chemistry continued. Absorption spectra of RuO_4 in 1M $HClO_4$ and of the red solution obtained by action of 4 equivalents of Hg_2^{++} on this solution are shown. Actions of various reducing agents on RuO_4 in 1M $HClO_4$ are summarized. A single-step procedure for the preparation of Ti(III) borohydride, consisting of the reaction of $TiCl_4$ with $LiBH_4$, is reported. Other studies reported briefly include phase studies on uranyl nitrate–H_2O–organic solvent systems, absorption spectrum of cobaltous nitrate in H_2O and organic solvents, Hf–Zr separation by fractional distillation, development of an atomic Cl apparatus and its use in preparation of the higher chlorides of transuranium elements, separation of Np and Pu chlorides by a flow method, determination of the capture cross section of Pu^{238} for thermal neutrons, and separation of T from H by thermal diffusion. (G.Y.)

13576 ANL-4326(Del.)
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-I. SUMMARY REPORT FOR APRIL, MAY, AND JUNE 1949. Aug. 3, 1949. Decl. with deletions Mar. 14, 1957. 37p. Contract W-31-109-eng-38. $6.30(ph OTS); $3.00(mf OTS).

13577 ANL-4469(Del.)
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-I SUMMARY REPORT FOR JANUARY, FEBRUARY, AND MARCH 1950. D. W. Osborne, ed. June 7, 1950. Decl. with deletions Feb. 12, 1957. 48p. Contract W-31-109-eng-38. $7.80(ph OTS); $3.30(mf OTS).

13578 ANL-5024(Rev.)
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-11 SUMMARY REPORT [FOR] OCTOBER, NOVEMBER, AND DECEMBER 1952. J. R. Gilbreath and O. C. Simpson, comps. Aug. 14, 1953. Decl. Mar. 9, 1957. 72p. Contract [W-31-109-eng-38]. $12.30(ph OTS); $4.50(mf OTS).

Progress is reported on the following investigations: effect of pile irradiation on the physical properties of graphite; the heat content and lattice constants of irradiated diamonds; lattice constants of irradiated BN; dimensional changes in irradiated silica and quartz; radiation-induced fragmentation of SiC; exposure and dosage for radiation damage experiments; the radiation-induced thermoluminescence of ice; absorption spectra of irradiated LiF; thermal release of charge carriers in x-irradiated LiF and thallium-ated KI; photocapacitive effect in alkali halide crystals; color centers in CaF_2; optical properties of irradiated diamond; effect of pile radiation on the decomposition of solutions of H_3BO_3 in light and heavy water; isotope effects in the γ-induced recombination of H_2 and H_2O_2; use of poisoned helices for reactor control; comparison of the radiation chemistry of light and heavy water solutions; oxidation of $FeSO_4$ in H_3BO_3 and Li_2SO_4 solutions by irradiation with thermal neutrons; spontaneous fission yields of Cm^{242}; fine structure in the molecular spectrum of T_2; Li isotopic determination by an optical spectrographic method; spectrographic determination of Hf in ZrO_2 by controlled atmosphere burning; determination of micro amounts of O_2 in U and Th; an improved continuous ether extractor for determination of U in dissolver solutions; and the 60-in. cyclotron. (D.E.B.)

13579 ANL-5039(Del.)
Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT FOR JANUARY, FEBRUARY, AND MARCH 1953. May 12,

1953. Decl. with deletions Feb. 12, 1957. 66p. Contract W-31-109-eng-38. $10.80(ph OTS); $3.90(mf OTS).

SEPARATION METHODS APPLIED TO HALIDE SYSTEMS. Studies of Ru decontamination from chloride solutions by coprecipitation with Bi showed that Mg was a most effective reducing agent. The possibility of extracting Nb out of a HCl solution of tetravalent U was indicated by distribution ratio studies using hexone as solvent. FLUORIDE VOLATILIZATION PROCESS STUDIES. Measurements of the rate of dissolution of U in BF_3 at 85°C were made. BiF_3, water, and aqueous KF_2 were not found effective in decontaminating Monel and Ni contaminated during the dissolution of irradiated U in BrF_3. Refractive indexes at 70°C were determined for various mixtures of BrF_5 and UF_6. U present in samples of STR-type fuel alloys was volatilized by treatment of the residue from concentrated HCl dissolution of the alloy with BrF_3 at 125°C. DEVELOPMENT OF ANALYTICAL METHODS. The feasibility of thiocyanate colorimetric methods for the determination of a variety of heavy metals which may be incorporated with fissionable material in reactor fuel alloys was demonstrated. A well-type scintillation counter was developed for use with liquid or solid samples. The processing of radioactive laboratory wastes is briefly outlined. (J.E.D.)

13580 ANL-5213
Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT FOR OCTOBER, NOVEMBER, AND DECEMBER 1953. Feb. 1, 1954. Decl. Feb. 12, 1957. 102p. Contract W-31-109-eng-38. $16.80(ph OTS); $5.70(mf OTS).

Progress is reported on the following investigations: Alterations of the process for EBR fuel recovery required by 2% Zr in fuel; pickling and dissolution of U–Zr alloys; partition of UO_2Cl_2 between 30 vol. % TBP in CCl_4 and aqueous HCl; electrolytic dissolution of U; droplet settling in systems of immiscible liquids; stacked stage extraction; rate of U–BrF_3 reactions; ignition experiments with the U–BrF_3 reaction; liquid-vapor equilibria for the system BrF_3–BrF_5; solid-liquid equilibrium for the BrF_5–UF_6 system; visible and ultraviolet absorption spectra of Br, BrF_3, and BrF_5; spectraphotometric determination of Br in BrF_3; separation of UF_6 from Br fluorides by sublimation; vacuum distillation; batch recrystallization, and zone melting; decontamination of molten U by slagging and volatilization, by electrorefining and by electrolytic fluorination; electrochemical measurements with quartz membrane cells; purification of U by zone melting; denitration of $UO_2(NO_3)_2$ in a fluidized bed; determination of F^-, Br^-, and low Pu in fluoride volatility samples; and processing and utilization of radioactive wastes. Procedures are given for the determination of F^- and low Pu in fluoride volatility samples. (W.L.H.)

13581 ANL-5254(Del.)
Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] JANUARY, FEBRUARY, AND MARCH 1954. May 1, 1954. Decl. with deletions Feb. 13, 1957. 123p. Contract W-31-109-eng-38. $19.80(ph OTS); $6.30(mf OTS).

Tentative flowsheets for the separation and decontamination of U and Pu in chloride solutions are given. The study of droplet settling and coalescence in immiscible liquid–liquid systems has been continued. The operation of the stacked stage extractor for the recovery of Pu was satisfactory. The rate of the U–BrF_3 reaction shows a small increase with increasing Br_2 concentration in the BrF_3. Absorption curves in the vapor phase are given for Br_2, BrF_3, BrF_5, and UF_6 over the range 210 to 800 mμ. Ni, Monel, K-Monel, and Inconel showed no corrosion greater than 0.08 mil/yr in pure BrF_3, 10% BrF_5–BrF_3, and 10% Br_2–BrF_3. A series of experiments on the fluorination of U–Pu alloys showed that no significant amount of Pu was converted to the volatile hexafluoride. Laboratory investigations of the reprocessing of reactor fuels at elevated temperatures are continuing along the lines of slagging of molten U, electrorefining of liquid Al–U alloys, electrolytic fluorination of U, and research in use of crystalline quartz cells for electrochemical measurements in fused salt systems. A continuous process for the conversion of $UO_2(NO_3)_2$ to UO_3 in a fluidized bed is being developed. A spectrophotometric method for the determination of fission-product Mo has been developed. Improved titrimetric determinations for W and Mo are described. Difficulties in the analysis for fluoride in volatility-process hydrolyzed samples have been corrected. Investigations on the behavior of exhausted cation resin beds and on the effectiveness of cationic resins for the removal of Sr showed that Sr did not start to leak until after hardness breakthrough. The fixation of activity of wastes in a form suitable for use as radiation sources is being studied. (T.R.H.)

13582 ANL-5602
Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION QUARTERLY REPORT [FOR] APRIL, MAY, JUNE 1956. Aug. 1, 1956. Decl. Feb. 18, 1957. 134p. Contract W-31-109-eng-38. $0.75(OTS).

Light scattering correlations were developed for 25% dispersions and several runs were made to investigate mass transfer in a continuous-flow mixing chamber using the system isobutanol–water. Fluoride volatilization separation processes were investigated. The production of UF_4 from ore concentrates by hydrogenation and hydrofluorination by fluidized-bed techniques are reported. The work on fluidization process development included the production of green salt from $UO_2(NO_3)_2$ solution, from solvent extraction refineries, and the production of UF_6 by direct fluorination of U ore concentrates. Studies were continued on reactor chemistry, chemical metallurgical separation processes, determination of U by x-ray spectrophotometric method, and radioactive waste processing. (J.E.D.)

13583 BMI-988
Battelle Memorial Inst., Columbus, Ohio.
STUDIES OF THE BOMB REDUCTION OF THORIUM HALIDES. H. A. Salier, J. R. Keeler, and L. J. Cuddy. Mar. 22, 1955. Decl. Mar. 1, 1957. 20p. Contract W-7405-eng-92. $0.30(OTS).

Limited studies were made of the bomb reduction of $ThCl_4$ with Ca and Mg to determine the feasibility of carrying out such reductions in the equipment presently being used for ThF_4 reduction. The bomb reduction of ThF_4 was also studied primarily to determine the feasibility of substituting Mg for all or part of the Ca reductant. Limited investigations of the ThF_4 reduction were also made to determine the effects produced by variation of the amount of $ZnCl_2$ and Ca in the charge, substitution of ZnF_2 for the $ZnCl_2$ booster, and addition of salts to the charge to reduce the viscosity of the slag. (auth)

13584 CF-52-1-52
Oak Ridge National Lab., Tenn.
IONIC CATALYSTS FOR THE RECOMBINATION OF HYDROGEN AND OXYGEN. C. H. Secoy. Jan. 8, 1952. Decl. Feb. 15, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

It is reported that variable-valence ions, including Cu, are capable of smoothly recombining H_2 and O_2. An activation energy of ~15 kcal. is involved, and for the recombination rate to be appreciable, the temperature must be in

the range 200 to 300°C. Recombination will occur whether the solution is under irradiation or not, whether the gas ($H_2 + O_2$) is supplied from cylinders or from the radiation decomposition of water. (L.M.T.)

13585 CF-56-5-193(Del.)
Oak Ridge National Lab., Tenn.
HEAT REMOVAL FROM ZIRCONIUM HYDROCHLORINATION. R. J. McNamee. May 22, 1956. Decl. with deletions Mar. 13, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

13586 CF-56-12-106
Oak Ridge National Lab., Tenn.
THE STABILIZATION OF AQUEOUS THORIUM NITRATE SYSTEMS AT 300°C WITH SULFATE. J. P. McBride. Dec. 26, 1956. Decl. Mar. 21, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Thorium nitrate solutions hydrolyze readily at 300°C to liberate HNO_3 and precipitate $Th(OH)_4$. The addition of trace quantities of sulfate to a $Th-U(NO_3)_4$ nitrate solution (U/Th mole ratio—0.005) has been found to markedly inhibit the hydrolysis at 300°C. It is postulated that a systematic investigation of acid deficient $Th(NO_3)_4$ solutions containing sulfate could result in the development of Th solutions or sols stable at 300°C and at concentrations to be of use in an aqueous homogeneous thorium breeder reactor. (auth)

13587 CN-1863
Clinton Labs., Oak Ridge, Tenn.
THE SOLUBILITY OF $BiPO_4$ IN HNO_3. Final Report on Problem Assignment No. 202-X59C—SOLUBILITY AND RATE OF SOLUTION OF $BiPO_4$ IN HNO_3. G. R. Leader. July 30, 1944. Decl. Apr. 4, 1957. 39p. Contract [W-7405-eng-39]. (A-2690). $6.30(ph OTS); $3.00(mf OTS).

Bismuth phosphate was shown to exist in at least two allotropic forms, differing in crystal structure and in solubility in HNO_3. As ordinarily precipitated under conditions of the $BiPO_4$ process, $BiPO_4$ comes down as hexagonal crystals, moderately soluble in 10N HNO_3. Upon prolonged digestion at 75° these hexagonal crystals are converted to a monoclinic structure only about 40% as soluble in 10N HNO_3. Additional data show $\alpha-BiPO_4$ solubility in HNO_3 to depend on the 2.35 power of the HNO_3 concentration at 25°C. Very nearly the same power dependence is also observed at 10, 50, and 75°C. Temperature is shown to have little effect on the solubility of $BiPO_4$ in the range 2 to 6N HNO_3, but at 8 and 10N HNO_3 a steady decrease in solubility with increasing temperature is observed. At 1N HNO_3, the solubility was found unchanged between 25 and 50°C, about 10% higher at 1° and about 20% higher at 75°C. The solubility of the β-form is found to show nearly the same power dependence upon HNO_3 concentration as does the α-form and is about 40% as great at a given acidity. The solubility of the β-form in 10N HNO_3 is shown to decrease with increasing temperature as does the solubility of the α-form. Studies indicated that nitrate ion plays an important part in the solution of $BiPO_4$ in HNO_3 because of the low degree of dissociation of $Bi(NO_3)_3$ in HNO_3 solutions. The solubility of $\alpha-BiPO_4$ in 10N HNO_3 containing various concentrations of HF was also determined. (auth)

13588 DPST-54-309
[Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.]
DEUTERIZATION RATE OF ION EXCHANGE RESINS. C. J. Banick. May 20, 1954. Decl. May 20, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

A batch experiment revealed that if maximum contact between D_2O and resin were maintained, as by stirring, complete equilibrium in the replacement of H_2O by D_2O would be reached within about 15 min. (auth)

13589 FMPC-131
National Lead Co. of Ohio, Cincinnati.
REVIEW OF AMES THORIUM PROCESS. E. W. Mautz and O. R. Magoteaux. [1953?]. Decl. Mar. 13, 1957. 49p. Contract AT(30-1)-1156. $7.80(ph OTS); $3.30(mf OTS).

A summary of the Ames Th process as reported in progress reports by NLO personnel during an extended training program at Ames to obtain background information for the startup and operation of the Fernald Th production plant is given. A general study of the Ames Th process included thorium oxalate precipitation, filtration, drying and calcination, hydrofluorination of ThO_2, reduction of ThF_4, Th–Zn alloy dezincing, Th metal casting, Ca distillation and preparation for bomb reduction, $ZnCl_2$ drying, analytical, spectrographic, and metallographic determinations, ZnF_2 drying for bomb reduction, and mesothorium consideration. (auth)

13590 HW-18128
[Hanford Works, Richland, Wash.]
PROPERTIES AND SETTLING RATES OF 101-U SLUDGE-SUPERNATE SLURRIES. R. B. Richards. June 20, 1950. Decl. Feb. 19, 1957. 4p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

13591 K-483
Carbide and Carbon Chemicals Corp. K-25 Plant, Oak Ridge, Tenn.
EFFECT OF URANIUM HEXAFLUORIDE ABSORPTION ON TENSILE PROPERTIES OF FLUOROTHENE. S. E. Frey, J. D. Gibson, and R. H. Lafferty, Jr. Sept. 2, 1949. Decl. Feb. 8, 1957. 15p. Contract [W-7405-eng-26.] $3.30(ph OTS); $2.40(mf OTS).

Fluorothene was subjected to immersion in liquid UF_6 at 62 to 70°C and at 100°C and subsequently tested for tensile properties and weight and dimensional changes. It was found that the various yield strengths of fluorothene were lowered by UF_6 absorption, but this lowering effect was diminished if the samples were allowed to stand after immersion prior to testing. The absorption of UF_6 by fluorothene was appreciable and was greatly facilitated by an elevation of temperature. The saturation limit of the UF_6 absorption by fluorothene at 100°C was found to be between 40 and 50% weight increase. The swelling of the fluorothene during absorption of UF_6 was readily correlated with the amount of UF_6 in the swollen plastic by a logarithmic expression. (auth)

13592 KAPL-M-MEB-1
Knolls Atomic Power Lab., Schenectady, N. Y.
PHYSICAL AND CHEMICAL PROPERTIES OF TBP. Mary E. Brennan. Mar. 9, 1951. Decl. Apr. 5, 1957. 15p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

This information was compiled from sources in the open literature and project reports. It is believed to be complete through 1950. (auth)

13593 KAPL-M-RNH-2
Knolls Atomic Power Lab., Schenectady, N. Y.
THE MANUFACTURE OF HAFNIUM METAL POWDER BY THE HYDRIDE PROCESS. R. N. Honeyman. Sept. 5, 1956. 5p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

13594 M-4562
Hooker Electrochemical Co., Niagara Falls, N. Y.
CONCENTRATION OF C-2 SLAG. W. G. Thomas. Feb.

17, 1944. Decl. Mar. 7, 1957. 9p. $1.80(ph OTS); $1.80 (mf OTS).

Screening C-2 slag with a 20-mesh screen and concentrating, by waste hydrochloric acid increased the metal content from 0.84 to 5.6%. The cost of the building and equipment for the concentration of 80 tons per month and the operating costs are included. (F.S.)

13595 MLM-368-1(Extract)
[Mount Lab., Miamisburg, Ohio].
BISMUTH COMPLEXES AT LOW HYDROCHLORIC ACID CONCENTRATIONS. J. H. Payne, Jr. [1949]. Decl. Mar. 27, 1957. 27p. $4.80(ph OTS); $2.70(mf OTS).

There is evidence of the formation of Bi^{3+}, $BiCl^{2+}$, $BiCl_2^+$, and $BiCl_4^-$ in the Cl^- concentration range 0.3 to 1.0M. Absorption spectra studies were made, and the extinction coefficients determined for each complex in this concentration range. (D.E.B.)

13596 NYO-1265
Hooker Electrochemical Co., Niagara Falls, N. Y.
ELECTROLYTIC BORON PROCESS. George T. Miller. May 31, 1956. Decl. Apr. 12, 1957. 30p. Contract AT(30-1)-1524. $9.30(ph OTS); $3.60(mf OTS).

A fused salt electrolytic method was developed for the production of elemental crystalline boron in 85% yield with a purity of 95%. An economic comparison with the Hot Wire process indicated a 30% reduction in production cost. (auth)

13597 ORNL-323(Pt.1)
Oak Ridge National Lab., Tenn.
TECHNICAL DIVISION-REPORT FOR QUARTER ENDING FEBRUARY 28, 1949. PART I. E. C. Miller, ed. Feb. 28, 1949. Decl. Mar. 15, 1957. 116p. Contract [W-7405-eng-26]. $18.30(ph OTS); $6.00(mf OTS).

The status of MTR Mockup design, testing, and construction is reviewed. Methods for control and removal of airborne particles are considered. Chemical engineering tests in solvent extraction, liquid metal heat transfer, coolant properties, and radiation effects are reported. The properties of some of the more important shielding materials were determined. (D.E.B.)

13598 TID-10145
New Brunswick Lab., AEC, N. J.
QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING SEPTEMBER 30, 1954. C. J. Rodden. Dec. 1954. Decl. Apr. 22, 1957. 50p. $0.45(OTS).

Various separate phases of the proposed moving bed reduction of UO_3 to UO_2 have been simulated in the moving bed reactor. The moving bed data indicated that UOx, even where x is as low as 2.35, was thermally dissociated in a steam gas phase with resultant liberation of stoichiometrically equivalent amounts of oxygen. Other sections of the reduction process investigated included the thermal dissociation of UO_3 feeds in the presence of steam, the H_2 reduction of U_3O_8 in the presence of steam, and H_2 reduction of UOx feeds in the presence of steam. The rate of thermal dissociation of $^3/_{16}$ inch UO_3 pellets to U_3O_8 was determined at 350°C. Specific laboratory rates of reduction of $^3/_{16}$ inch U_3O_8 pellets using 3.5 mole % H_2 in steam at 460, 550, 575, and 625°C were found to vary from 0.072% conversion per minute at the lower temperature to 3.08 at the higher temperature. At low concentrations of HF the hydrofluorination of $^3/_{16}$ inch UO_2 pellets in a given gas phase over a wide temperature range indicated that a region of maximum specific reaction rates existed for each gas phase. Setting times of UO_3-H_2O mixtures were studied as a function of setting temperatures and wt. % H_2O. Shear strengths of the resulting $^3/_{16}$ inch pellets were obtained in the hydrated, dehydrated, and reduced states. Regardless of the conditions of preparation investigated, the shear strength of the hydrated pellets appeared to reach a maximum at from 10 to 20 hours after preparation and then decreased to a value of from 8 to 14 pounds upon standing for several days. A Leitz Dilatometer was used to investigate the dimensionable stability of rolled Th metal. Thermal diagrams were obtained for maximum temperatures of 720, 825, and 920°C. Grain growth begins at 875°C and at 920°C the growth is rather pronounced. (auth)

13599 TID-10146
New Brunswick Lab., AEC, N. J.
QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING MARCH 31, 1955. C. J. Rodden. June 1955. Decl. Apr. 1, 1957. 56p. $0.50(OTS).

Synthetic UO_3 mixtures, containing various amounts of B either as H_3BO_3 or Na_2BO_3, were prepared. This material was treated with H_2 and then with HF at 500°C in a laboratory furnace. Starting with 850 ppm of B in the UO_3, UF_4 containing as little as one part per million of B has been produced. The measurement of pH to 0.01 units can be used as a means of evaluating the acidities of uranium process streams with an accuracy of 2 to 5%. Distribution coefficients (organic to aqueous) of V^{5+} in the ether-HNO_3-H_2O system were found for 3 and 5 normal HNO_3 to be 0.004 and 0.016, respectively. Samples of Th from two ingots were analyzed by chemical, spectrochemical, and metallographic methods for composition, inclusion content and grain size. The chemical composition and spectrographic results are given. The study of the determination of U by derivative polarography has been continued. A survey was made of the effect of Cu, Bi, Fe, Ni, Zn, Co, Pb, V, and Mo in a 0.25 M H_2SO_4 medium. A 0.25 M Na_2SO_4 medium was also studied for its applicability to the derivative polarography of U. The feasibility of scooping instead of weighing MgO charge for spectrochemical analysis was investigated. A variation in the weight of sample charge was not found to affect the analytical results appreciably. A spectrochemical procedure for the determination of Fe in B metal has been developed employing the porous cup technique. (auth)

13600 WAPD-137
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
ZIRCONIUM-WATER REACTIONS. B. Lustman. Dec. 1, 1955. Decl. Mar. 18, 1957. 42p. Contract AT-11-1-GEN-14. $0.60(OTS).

The available data on zirconium-water reactions were applied to determine the course and extent of reaction between zirconium-base alloys and water under conditions corresponding to a loss of coolant accident in the Pressurized Water Reactor. The analysis consisted of the determination of the heating curve of Zircaloy-2 while in the core structure under the influence of heat released in the chemical reaction plus residual decay heat, and the determination of the cooling curve of a molten droplet as it drops into water. In addition, a physical picture was proposed of the distribution of energy and products of the chemical reaction. The analysis revealed that less than 20% reaction would be expected under the postulated conditions. The amount of reaction within the core structure was quite sensitive to even minor rates of cooling. The extent of reaction of molten metal droplets with water was very sensitive to the emissivity of the surface, and less sensitive to particle size and residual decay heat. (auth)

13601 WAPD-CP-79
[Westinghouse Electric Corp.] Atomic Power Div., Pittsburgh.
LITHIUM BORATE SOLUBILITY TEST IN A DYNAMIC

SYSTEM. B. G. Schultz. Sept. 3, 1953. Decl. Mar. 19, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

From observation of the data, it is concluded that hideout of H_3BO_3 in the loop is not a result of temperature differentials or the hydrogen concentration, but a concentration of the solution in the surge tank due to evaporation of the solvent. It is concluded from this test that no hideout or preferential deposition of H_3BO_3 will occur in the Mark I system since there is no place at which concentration by evaporation can take place except in the degassifier and there the direction of flow does not permit it. (auth)

13602 Y-348(Del.)
Carbide and Carbon Chemicals Corp. Y-12 Plant, Oak Ridge, Tenn.
PURIFICATION OF CALCIUM BY DISTILLATION SEMI-WORKS DEVELOPMENT. W. M. Leaders, C. C. Haws, and G. B. Marrow. Feb. 23, 1949. Decl. with deletions Mar. 5, 1957. 32p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

The method developed through these experiments involves the maintenance of proper temperature-pressure relations so as to secure refluxing near the still-pot. The cold finger temperature determines the physical properties of the distillate. A high efficiency is attained in production of Ca of acceptable quality. (T.R.H.)

ANALYTICAL PROCEDURES

13603 BMI-98
Battelle Memorial Inst., Columbus, Ohio.
BETA-COUNTING METHODS APPLIED TO THE DETERMINATION OF URANIUM IN LOW-GRADE ORES. J. R. Doig, C. M. Schwartz, Iver Igelsrud, and H. R. Nelson. July 1947. Decl. Feb. 15, 1957. 76p. Contract W-38-094-eng-27. $12.30(ph OTS); $4.50(mf OTS).

The application of the β counter as an analytical tool useful in experiments on the extraction and concentration of U from low-grade shale and phosphate rock ores was investigated and found to be satisfactory for samples which have received gravity separation and flotation treatments. For samples given pyrometallurgical or other chemical treatments the β counter yields erroneous results for U concentration because the secular equilibrium of the sample is disturbed. (F.S.)

13604 CF-56-4-101
Oak Ridge National Lab., Tenn.
HRP-CP: ANALYTICAL REQUIREMENTS FOR THE HRT CHEMICAL PROCESSING PLANT. William L. Carter. Apr. 9, 1956. Decl. Mar. 18, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The proposed analytical requirements for the Homogeneous Reactor Test chemical processing plant are tabulated for both infrequent and routine analyses conducted for determining fission product, corrosion product, off-gas, and fuel component concentrations in fuel and blanket systems. (T.W.S.)

13605 DP-136
Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.
NONDESTRUCTIVE MEASUREMENT OF THE URANIUM CONTENT OF URANIUM–ALUMINUM ALLOYS. L. E. Goodwin. Oct. 1955. Decl. Feb. 20, 1957. 15p. Contract AT(07-2)-1. $0.30(OTS).

An instrument is described which measures the U content of a 0.080-inch thick plate of Al–U alloy. The content is measured by γ transmission to an accuracy of ± 0.0006 g/cm^2 over the range of 0.085 to 0.115 g/cm^2 of contained U. (auth)

13606 LA-1957
Los Alamos Scientific Lab., N. Mex.
THE DETERMINATION OF VANADIUM AND CHROMIUM IN ALLOYS WITH URANIUM. Hobart H. Willard and Owen H. Kriege. Sept. 1955. Decl. Apr. 1, 1957. 22p. Contract W-7405-eng-36. $0.35(OTS).

Two methods are described for the quantitative determination of vanadium in uranium–vanadium alloys. In the first method the alloy is dissolved in hydrochloric acid, with the addition of hydrogen peroxide, and boiled to destroy the latter. The vanadium is oxidized by adding an excess of ceric sulfate, the excess ceric sulfate is reduced by adding sodium azide, and the vanadic acid is titrated potentiometrically with standard ferrous sulfate. A procedure is also described for the quantitative determination of vanadium and chromium in uranium–chromium–vanadium alloys. The vanadium and chromium are oxidized to vanadic and chromic acids by boiling with perchloric acid. The solution is quickly cooled, diluted, and titrated potentiometrically with standard ferrous sulfate, both the vanadium and chromium being reduced. The vanadium is oxidized again by an excess of ceric sulfate, the excess of which is removed by adding sodium azide. The vanadic acid is titrated potentiometrically with ferrous sulfate. The difference between the two titrations is the amount of ferrous sulfate equivalent to the chromium. An average of 99.88% of the vanadium present was found with an average standard deviation of 0.37%. For chromium an average of 100.03% of the amount present was found with an average standard deviation of 0.33%. (auth)

13607 KAPL-M-JR-6
Knolls Atomic Power Lab., Schenectady, N. Y.
PROCEDURES FOR THE CHEMICAL ANALYSIS OF 7% URANIUM–ZIRCONIUM AND 7% URANIUM–ZIRCALOY. J. Rynasiewicz, comp. May 31, 1956. 21p. Contract [W-31-109-eng-52]. $3 30(ph OTS); $2.40(mf OTS).

13608 LA-1948
Los Alamos Scientific Lab., N. Mex.
THE DETERMINATION OF SMALL AMOUNTS OF THORIUM IN THE PRESENCE OF URANIUM, TITANIUM, AND TUNGSTEN. Hobart H. Willard, Arthur W. Mosen, and Ross D. Gardner. June 1955. Decl. Apr. 1, 1957. 15p. Contract [W-7405-eng-36]. $3.30(ph OTS); $2.40(mf OTS).

Thorium was determined in uranium alloys, containing 0.1 to 3.0% Th, by separation as the fluoride followed by a complexometric titration. An average recovery of 99.4% ($\sigma = \pm 0.56\%$) was obtained. Tungsten, Ti, Nb, Zr, Fe, and Ta in the alloys did not interfere. (auth)

FLUORINE AND FLUORINE COMPOUNDS

13609 CF-52-3-230
Oak Ridge National Lab., Tenn.
DENSITIES OF CERTAIN SALT MIXTURES AT ROOM TEMPERATURE. M. Tobias, S. I. Kaplan, and S. J. Claiborne. Mar. 26, 1952. Decl. Feb. 20, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

13610 CF-55-3-62
Oak Ridge National Lab., Tenn.
MEASUREMENT OF THE VISCOSITY OF COMPOSITION 30. S. I. Cohen and T. N. Jones. Mar. 9, 1955. Decl. July 5, 1957. 3p. Contract [W-7405-eng-26]. $1.80 (ph OTS); $1.80(mf OTS).

Viscosity measurements were made on Composition 30 ($NaF-ZrF_4-UF_4$; 50–46–4 mole %), and the viscosity was found to vary from about 11 centipoises at 570°C to about

3.4 centipoises at 870°C. The viscosity may be represented throughout this temperature range by μ = 0.1307 exp (3730/T) (where T is in °K) within an accuracy of 12%. (L.T.W.)

13611 CF-55-8-21
Oak Ridge National Lab., Tenn.
MEASUREMENT OF THE VISCOSITY OF COMPOSITION 88. S. I. Cohen and T. N. Jones. Aug. 1, 1955. Decl. Feb. 20, 1957. 3p. Contract [W-7405-eng-26]. $1.80 (ph OTS); $1.80(mf OTS).

Viscosity measurements were made on Composition 88 ($NaF-LiF-BeF_2$; 64-5-3 mole %) and the viscosity was found to vary from about 8 centipoises at 580°C to about 2.7 centipoises at 850°C. (auth)

13612 CF-55-9-31(Del.)
Oak Ridge National Lab., Tenn.
MEASUREMENT OF THE VISCOSITY OF COMPOSITION 70. S. I. Cohen and T. N. Jones. Sept. 6, 1955. Decl. with deletions Mar. 15, 1957. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Viscosity measurements were made on Composition No. 70 ($NaF-ZrF_4-UF_4$; 56-39-5 mole %), and the viscosity was found to vary from about 8.1 centipoises at 600°C to about 3.6 centipoises at 800°C. The viscosity may be represented throughout the temperature range by μ = 0.1045 exp(3798/T), where T is in °K. (L.T.W.)

13613 CF-55-11-28
Oak Ridge National Lab., Tenn.
MEASUREMENT OF THE VISCOSITY OF COMPOSITIONS 90 AND 96 AND ($KF-BeF_2$; 79-21 Mole %). S. I. Cohen and T. N. Jones. Nov. 8, 1955. Decl. Mar. 16, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The viscosity of Composition 90 ($NaF-KF-BeF_2$; 49-15-36 mole %) varied from about 8.0 centipoises at 600°C to about 2.9 centipoises at 850°C and may be represented throughout this temperature range by μ = 0.0837 exp(3982/T), where T is in °K. The viscosity of Composition 96 ($NaF-LiF-BeF_2$; 53-24-23 mole %) varied from about 5.9 centipoises at 600°C to about 3.0 centipoises at 800°C and may be represented by μ = 0.0185 exp(3018/T). The viscosity of $KF-BeF_2$ (79-21 mole %) was also measured from 750 to 850°C, the viscosity range being 2.6 to 1.9 centipoises, respectively. The viscosity relationship is μ = 0.0777 exp(3590/T). (L.T.W.)

13614 CF-56-5-67
Oak Ridge National Lab., Tenn.
HEAT CAPACITY OF COMPOSITION NO. 100. W. D. Powers and G. C. Blalock. May 14, 1956. Decl. July 5, 1957. 2p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The enthalpy and heat capacity of Composition 100 (LiF-NaF; 60-40 mole %) were determined in the liquid and solid state. The heat of fusion was also calculated. (L.T.W.)

13615 CF-56-5-68(Del.)
Oak Ridge National Lab., Tenn.
HEAT CAPACITY OF COMPOSITION NO. 81. W. D. Powers and G. C. Blalock. May 14, 1956. Decl. with deletions Mar. 15, 1957. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The enthalpy of Composition No. 81 (LiF-NaF-ZrF; 55-22-23 mole %) was determined in the temperature range 100 to 900°C. There are two distinct discontinuities in the temperature enthalpy relationship, one at 370 to 380°C and the other at 565 to 585°C. The lower temperature discontinuity is assumed to be a phase transition from a β form stable below 375°C to an α form stable above 375°C. The discontinuity near 570°C is the fusion of the β form. The enthalpies, heat capacities, and heats of transition of the three forms were calculated. (L.T.W.)

GRAPHITE

13616 BMI-1042(Del.)
Battelle Memorial Inst., Columbus, Ohio.
EFFECT OF CONSTITUTION OF GRAPHITES ON THEIR STABILITY UNDER LOW-TEMPERATURE IRRADIATION. Luther D. Loch, Alfred E. Austin, Ralph J. Harrison, and Winston H. Duckworth. Sept. 30, 1955. Decl. with deletions Feb. 13, 1957. 74p. Contract W-7405-eng-92. $2.00(OTS).

Research on the relationships between graphite constitution and irradiation stability resulted in the development of graphite types with greatly improved dimensional stability under cold test-hole irradiation. The constitution of a graphite was characterized by x-ray determinations of the size, perfection, and unit-cell dimensions of its crystallites, by helium-gas and organic-liquid displacement measurements of its microporosity, and by mercury porosimetry measurements of the macropores. Other bulk properties and differences in fabrication technique also were investigated. The bulk irradiation expansion shown by a graphite was found to depend upon the amount of intercrystalline space available for accommodation of crystal expansion. Bulk expansion correlated well with micropore volume directly or with the ratio δ/L_C, where L_C is the crystallite dimension perpendicular to the layers. The same correlations applied to bulk thermal expansion, thus explaining the empirical relationships that have been observed between thermal expansion and irradiation expansion. The resin carbon and sugar carbon graphites had the largest micropore volumes and contracted instead of expanding. The contractions appear to result from the combined effects of the a_0 lattice contraction and an irradiation-annealing effect which tends to decrease the average spacing between crystallites. The best dimensional stability was shown by an experimental graphite made from air-blown petroleum asphalts, and designated as Graphite 36. This material contracted slightly up to about 750 MD/CT and then expanded at a low rate. (auth)

13617 CC-976
Chicago. Univ. Metallurgical Lab.
GRAPHITE PURITY RESEARCH PROGRAM. G. E. Boyd, R. E. Curtis, and W. H. Johnston. Dec. 22, 1943. Decl. Apr. 2, 1957. 70p. Contract W-7401-eng-37. (A-1652). $10.80(ph OTS); $3.90(mf OTS).

Chemical analytical studies of graphite raw materials have revealed new sources of petroleum cokes and pitches of greater purity than those previously used in the manufacture of project graphite. The use of these raw materials of exceptional purity has resulted in an improvement in k of the order of one per cent as compared with the AGOT-AGNT graphite typical of the previous production. A study of the graphitizing process has shown that no dangerous impurities are introduced into the graphite in manufacture; rather, the process actually purifies. Furthermore, impurities are distributed homogeneously within the furnace charge; thus, the AGOT-AGNT distinction is no longer necessary. A comparison of the methods of graphite testing has demonstrated excellent correlations between chemical analytical data, the results of the functional tests at Argonne, and the sigma pile experiments. It may therefore be inferred that chemical testing should be an adequate control of the graphite purity. Methods are described for

the analysis of ash, B, V, Ti, Fe, and Ca in petroleum coke, pitch, and graphite. (auth)

13618 HW-45035

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

GRAPHITE DIFFUSION LENGTH MEASUREMENTS AT HANFORD. C. R. Richey and E. Z. Block. Sept. 10, 1956. Decl. Feb. 23, 1957. 41p. Contract W-31-109-Eng-52. $0.30(OTS).

A series of diffusion length measurements were carried out on graphite stacks of various constructions in an attempt to resolve the discrepancies between the graphite diffusion lengths measured in the Hanford reactors and that value measured in the Hanford Standard Pile. It was found that the diffusion length of the graphite in the Hanford reactors is in good agreement with the value for the Hanford Standard Pile when corrections are made for the absorption and scattering of neutrons by the Al process tubes, and for neutrons streaming in the channels of the reactors. (auth)

13619 NAA-SR-146

North American Aviation, Inc., Downey, Calif.

A THEORETICAL TREATMENT OF ELECTRONIC PHENOMENA IN A TWO-DIMENSIONAL GRAPHITE LATTICE. W. P. Eatherly. July 25, 1951. Decl. Feb. 27, 1957. 29p. Contract AT-11-1-GEN-8. $0.35(OTS).

The general theory of conduction phenomena is reviewed and developed in tensor form. The resulting relations are then used to compute the electrical conductivity, the Hall coefficient, the magneto-resistivity, and the magnetic susceptibility of graphite. It is assumed that the anistropy is sufficient to permit approximating graphite by a two-dimensional lattice, and that the mean free path of the excited electrons is a constant. The resulting expressions are shown to agree satisfactorily with experimental data on radiation damage and annealing. The theory is criticized and the extension to three dimensions discussed. It is concluded that present calculations are not sufficiently fine to permit construction of a detailed mechanism, but that increased scattering and electron trapping do account for the variation of known electronic phenomena with radiation damage. More refined computations are being explored but are incomplete. (auth)

13620 NAA-SR-194

North American Aviation, Inc., Downey, Calif.

THE DIFFUSION OF XENON FROM URANIUM CARBIDE-IMPREGNATED GRAPHITE AT HIGH TEMPERATURES. D. Cubicciotti. Oct. 13, 1952. Decl. Mar. 28, 1957. 40p. Contract AT 11-1-GEN-8. $0.35(OTS).

The rate of escape of radioactive Xe from neutron-irradiated U-impregnated graphite was investigated in the temperature range 1000 to 1600°C and found to be independent of the sample size. The fraction of Xe that escapes can be expressed by the diffusion equation. The diffusion path is through the grains of the graphite. An estimate of the Xe^{135} that escapes at steady state without capturing neutrons shows that at 1500°C only 12% of the Xe^{135} produced escapes without neutron capture and that at 2000°C about 56% is expected to escape. (F.S.)

LABORATORIES AND EQUIPMENT

13621 CF-52-2-214(Del.)

Oak Ridge National Lab., Tenn.

DEVELOPMENT AND OPERATION OF A SEMI-CONTINUOUS SYSTEM FOR THE PRODUCTION OF FERROUS SULFAMATE. W. B. Watkins. Feb. 26, 1952. Decl. with deletions Feb. 13, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A system designed to produce semi-continuously ferrous sulfamate of a quality acceptable to the Purex process has been installed in the ORNL pilot plant and successfully operated. It has been demonstrated that approximately 18 gallons of ferrous sulfamate having a reducing normality of 1.2 can be produced in 18 to 20 hours. Thirty-five pounds of technical grade sulfamic acid, about 10 pounds of 1-in. iron wire nails and 200 ft^3 of nitrogen (purge gas) are required to produce one 18 gallon charge. A minimum of operator attention (about one man hour per charge) is required and no difficulty has been encountered in filtering the final solution. Quantities of ferrous sulfamate produced by the semi-continuous system have been employed in ORNL pilot plant runs at full Hanford activity level and have been entirely satisfactory. (auth)

13622 CF-56-5-131

Oak Ridge National Lab., Tenn.

HRP-CP: DESIGN CALCULATIONS ON SUPERHEATERS H-7 AND H-8 FOR HRT CORE PROCESSING PLANT. William L. Carter. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The HRT core processing plant is provided with a superheater to serve each of the decay-storage tanks in the low pressure system. These units raise the temperature of a D_2O vapor, D_2 and O_2 gas stream considerably above the dew point of the heavy water to preclude condensation in the system. Since the gas stream is being routed to a catalytic recombiner for D_2 plus O_2 recovery, it is necessary to ensure having a dry catalyst bed. Also it is desirable to maintain the gas at a temperature level that will give a high recombination percentage. The superheaters are jacketed sections of the process vapor line and are heated by 250 psig steam. Pertinent design and operating data for the units are tabulated. (auth)

RADIATION CHEMISTRY

13623 ANL-4999

Argonne National Lab., Lemont, Ill.

THE EFFECT OF IONIZING PARTICLES ON LIQUID WATER AND AQUEOUS SOLUTIONS. Edwin J. Hart. Mar. 6, 1953. Decl. Feb. 23, 1957. 20p. Contract W-31-109-Eng-38. $0.15(OTS).

The principles of the interaction of ionizing radiations with water are discussed in detail with the radiation chemistry of water explained by the chemistry of hydrogen and hydroxyl free radicals and their reactions with impurities. Neutron and γ dosimetry and water decomposition in the Argonne Heavy Water Reactor are considered also. (F.S.)

13624 KAPL-M-PHK-2

Knolls Atomic Power Lab., Schenectady, N. Y.

REMARKS ON THE INTERACTION OF RADIATION WITH THE WALLS OF VESSELS AS IT AFFECTS THE RADIATION CHEMISTRY OF THEIR CONTENTS. Philipp H. Klein. May 8, 1956. 8p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

RADIATION EFFECTS

13625 BMI-X-129

Battelle Memorial Inst., Columbus, Ohio.

STATUS OF AN ALTERNATE CONTROL MATERIALS PROGRAM. S. J. Paprocki. July 15, 1956. Decl. Feb. 16, 1957. 14p. $3.30(ph OTS); $2.40(mf OTS).

The radiation resistance of B, BC, and ZrB_2 powders was investigated. Fragmentation due to radiation was not ob-

served generally. The irradiated powders appeared to be extremely hygroscopic at room temperature. (C.W.H.)

13626 CC-2099

[Chicago. Univ. Metallurgical Lab.]

EFFECT OF RADIATION ON CHROMATE IN W WATER. III. INHIBITION EXPERIMENTS IN THE CP-3 PILE. M. G. Bowman, M. Burton, S. G. Davis, J. A. Ghormley, and S. Gordon. Sept. 2, 1944. Decl. Feb. 19, 1957. 12p. $3.30(ph OTS); $2.40(mf OTS).

The effect of various inhibitors on the production of Cr^{+3} in W waters containing chromate has been studied at a variety of pH's in the CP-3 pile. The addition of 2 ppm $NaClO_3$ appears to have little effect at a pH of 6.3, but 2 ppm $NaClO_3$ or ~0.15 ppm of Cl_2 (in the form of NaOCl) somewhat decrease the amount of chromate reduced at a pH of 7.1. The inhibition is not great enough, however, to reduce the amount of Cr^{+3} formed to the maximum concentration permission at W (0.02 ppm). The addition of 2 ppm $(NH_4)_2S_2O_8$ as an inhibitor will permit W water to be used at a pH of 7.0, but not at a pH of 6.5 as far as chromate reduction is concerned. The concentration of Cr^{+3} would be ~0.01 ppm and ~0.05 ppm, respectively, after one passage through the pile. Previous conclusions as to the effects of temperature, pH, intensity, and the addition of ammonium persulfate, which had been reached from experiments with simulated W water, have been checked by studies with solutions made from water shipped directly from Hanford. All of the effects noted were found to be similar to those already reported, but the actual chromate reduction appears to be slightly less in the real W water sample studied than in simulated W water. (auth)

13627 CF-50-6-26(Del.)

Oak Ridge National Lab., Tenn.

RADIATION DAMAGE. Chapter IV, Part 1. G. E. Evans. June 7, 1950. Decl with deletions Mar. 20, 1957. 21p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Data are presented from studies on the effects of radiation on reactor materials. Topics discussed include the design of reactor fuel elements, and radiation effect on fissionable materials, uranium-bearing materials, BeO–UO_2 compacts, and graphite. (C.H.)

13628 CF-56-5-56

Oak Ridge National Lab., Tenn.

A PRELIMINARY STUDY OF ULTRAVIOLET SPECTRA FOR DETECTION OF SOLVENT DAMAGE BY RADIATION. J. R. Flanary, A. H. Kibbey, and V. L. Tweedie. May 7, 1956. Decl. Mar. 13, 1957. 22p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

A preliminary study was made of the use of ultraviolet spectra for detection of radiation-induced chemical change in Special Naphtha No. 1, a refined kerosene recommended as diluent for tributyl phosphate in highly radioactive solvent-extraction processes. The ultraviolet spectrum was more sensitive to chemical change than the usual chemical tests. Unsaturation produced by radiation was indicated by ultraviolet spectra and by iodine number and chromyl chloride reduction; however, the data were insufficient to establish correlations of spectra and chemical tests. The maximum irradiation level was 595 β watt-hr/liter of solvent. Results of batch extraction and stripping tests with irradiated Special Naphtha No. 1 and tributyl phosphate are presented. (auth)

13629 KAPL-M-GEG-8

Knolls Atomic Power Lab., Schenectady, N. Y.

TYPE AND QUANTITY OF GASES EVOLVED BY GAMMA IRRADIATION OF GLASS-INSULATED AND PLASTIC-INSULATED ELECTRICAL POWER CABLES. George E. Galonian. Aug. 8, 1956. 22p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

The results of an investigation to determine the type and quantity of gas evolved from gamma irradiation of glass-and-plastic-insulated electrical cables designed as power cables for electromagnetic pumps are reported. The principal gases evolved are hydrogen and methane after 10^7r and ethane after 10^8r total gamma exposure of silicone-and-glass-insulated cables. Small amounts of benzene are produced after 10^9r total gamma dosage. Benzene and butane are also evolved from plastic-insulated cables. The quantity of gas increases exponentially (exponent less than one) with increasing gamma dosage and amounts to about 1000 cc at NTP for silicone-and-glass-insulated cables after 10^9r and about 100 cc at NTP for plastic-insulated cables after 10^8r. (auth)

13630 M-3796(Rev.)

Clinton Labs., Oak Ridge, Tenn.

PILE TECHNOLOGY LECTURE 34. EFFECTS OF RADIATION ON MATERIALS—CHEMICAL COMPOUNDS. A. O. Allen. [194?]. Decl. Mar. 6 1957. 13p. $0.25(OTS).

The effects of radiation on covalently bonded compounds is discussed with emphasis on materials associated with reactors. (T.R.H.)

RARE EARTHS AND RARE-EARTH COMPOUNDS

13631 KAPL-M-WKA-11

Knolls Atomic Power Lab., Schenectady, N. Y.

AN ESTIMATE OF THE COST OF EUROPIUM OXIDE FROM MONAZITE. W. K. Anderson and F. D. Leipziger. Apr. 10, 1956. 15p. Contract W-31-109-Eng-52. $3.30 (ph OTS); $2.40(mf OTS).

The cost of europium oxide production as a function of ore enrichment and rate of production are examined. For ore enrichments between 0.05% and 0.1% and for production rates of the order of 1000 kg/yr, the cost would be expected to lie in the neighborhood of $2 to $3/gram where europium is the sole product. Normal by-product relationships with other desirable materials occurring in the monazite, should quickly bring this cost to below $1/gram. (auth)

SEPARATION PROCEDURES

13632 A-1069

Yale Univ., New Haven. Sterling Chemistry Lab.

A SUPPLEMENTARY REPORT ON THE DESIGN, OPERATION AND EFFICIENCY OF A COUNTERCURRENT LIQUID-LIQUID SIEVE-PLATE EXTRACTOR FOR THE ETHYL ETHER EXTRACTION OF URANYL NITRATE. S. H. Brown, Herbert H. Clark, A. L. Grunewald, Edward J. King, J. L. Miller, Ralph G. Van Name, E. G. Robles, R. S. Smith, and Scott E. Wood. Jan. 24, 1945. Decl. Mar. 14, 1957. 10p. $1.80(ph OTS); $1.80(mf OTS).

This report supplements A-1062.

Three problems of design and operation, the reuse of ether, use of commercial ether, and the through-put capacity of the sieve-plate extractor are studied with reference to a countercurrent liquid-liquid extraction of uranyl nitrate. (F.S.)

13633 ANL-4545(Del.)

Argonne National Lab., Lemont, Ill.

CHEMISTRY DIVISION, SECTION C-1 SUMMARY REPORT FOR JULY, AUGUST, AND SEPTEMBER 1950. D. W. Osborne, ed. Dec. 11, 1950. Decl. with deletions Feb. 12, 1957. 36p. Contract W-31-109-eng-38. $6.30(ph OTS); $3.00(mf OTS).

The half-life of Tc^{99} has been determined to be 2.12×10^5 yr ± 2% error. Activities are also reported concerning the following: the tetravalent and pentavalent states of Pa; separation of Pa and U from neutron irradiated ionium; separation of Pa and Th by chloride distillation; phosphorus oxychloride-uranium chloride complexes; solubility of inorganic fluorides in Halogen fluorides; dissolution of U metal; decontamination experiments; U-O system; Np (V) compounds; electro-chromatographic separations; heats of solution of uranyl nitrate hydrates in various solvents; fractionation of rare earths using a hydrochloric acid-mixed butyl phosphate system; and extraction of Am and Cm into butyl phosphate solvents from various aqueous phases. A discussion of a method of separation means of a butyl phosphate-aqueous citrate system are included. (D.E.B.)

13634 BMI-243A
Battelle Memorial Inst., Columbus, Ohio.
RECOVERY OF THORIUM AND URANIUM FROM MONAZITE SAND. VOLUME II. Final Report. G. D. Calkins, R. B. Filbert, Jr., A. E. Bearse, and J. W. Clegg. Sept. 15, 1950. Decl. May 3, 1957. 173p. Contract AT-30-1-gen-228. $1.10(OTS).

This is a companion report to Volume I (BMI-243). Details are presented on the processing of monazite sand and on separating the rare earths from thorium and uranium. Solvent-extraction and ion-exchange techniques for purifying thorium and uranium are described. (auth)

13635 BMI-967
Battelle Memorial Inst., Columbus, Ohio.
THE DISTRIBUTION OF THORIUM-DECAY PRODUCTS IN THE THORIUM-PURIFICATION PROCESS. G. D. Calkins, M. Poberskin, J. E. Gates, R. H. Blackmore, and J. H. Duffee. Dec. 7, 1954. Decl. Mar. 1, 1957. 46p. Contract W-7405-eng-92. $0.40(OTS).

The distribution of thorium-decay products in the thorium-purification process used at the Feed Materials Production Center was investigated. Decontamination of the thorium during fluoride precipitation was insufficient to eliminate the health hazard at later stages of the process. Solvent extraction was very efficient in decontaminating the thorium, but the gross radioactivity regrowth was relatively rapid. Process wastes of the fluoride precipitation and solvent extraction were readily decontaminated by carrier precipitation with barium sulfate and subsequent neutralization. Precipitation of thorium as the oxalate resulted in decontamination of the thorium from approximately 90% of its decay-product radioactivity. (auth)

13636 CF-50-2-27(Del.)
Oak Ridge National Lab., Tenn.
TENTATIVE PLAN FOR THE RECOVERY OF FISSION PRODUCTS AND NEPTUNIUM FROM REDOX SPECIAL RUNS. G. W. Parker. Feb. 8, 1950. Decl. with deletions Feb. 13, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The recovery of gram quantities of long-lived radioactive and stable fission products and Np^{237} from the 1AW raffinate solutions (Redox Process) is discussed. (C.W.H.)

13637 CF-52-1-189
[Oak Ridge National Lab., Tenn.]
A PRECIPITATION PROCESS FOR THE SEPARATION OF RADIOBARIUM FROM IRRADIATED MTR ASSEMBLIES. Quarterly Report [For] November 1951 Through January 1952. J. H. Pannell. Jan. 31, 1952. Decl. Apr. 2, 1957. 15p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

Separation of barium by precipitation has been tested as an alternative to ion exchange in the RaLa Process. The removal of Sr, Ce, and Al have been studied under optimum conditions for recovery and decontamination of Ba. (auth)

13638 CF-52-8-136
Oak Ridge National Lab., Tenn.
PULSE GENERATOR DEVELOPMENT. Quarterly Report for Period May–July 1952. E. Lieberman and A. C. Jealous. Aug. 13, 1952. Decl. Feb. 14, 1957. 16p. Contract [W-7405-eng-26]. $10.80(ph OTS); $3.90(mf OTS).

Further development of the semi-square pulse wave was delayed until the mechanical pulser with the special cam could be made available for column tests. Some tests along this line are contemplated during the August–October, 1952 quarter. Large pulse columns will require pulse generator drives in the range of 50 to 150 HP if liquid-liquid systems with high average specific gravity and pulse frequencies up to 200 cycles/min are incorporated. Cyclically varying pressure conditions in such cases will necessitate special pumping considerations. The mechanical pulse generator, incorporating a special cam to provide a pulse cycle incorporating a high speed pressure stroke, pause, and slow return stroke, was satisfactorily tested for equipment operability and reliability during a continuous 30-day test period. Leakage rate of the 5 in. diameter Graphitar piston against a 20 ft. water head was less than 10 cc/min. A hydraulically-driven pulse generator, incorporating a standard Vickers Company hydraulic control system which permits a wide variation in pulse conditions was operated continuously for over 80 days without variation in pulse conditions or any other troubles. Leakage rate of the 6 in. diameter Graphitar piston operating against a 20 ft. water head dropped during the test period from 30 to 15 cc/minute. Four mechanical-drive pulse generators for 6 to 10 in. diameter columns and incorporating variable frequency and amplitude, but providing only sine wave operation, are now being tested prior to installation on the pulse columns in Bldg. 3505. A piston leakage rate of about 5 cc/minute was noted during a 7 day test on the first unit tested with a 20 ft water heat. (auth)

13639 CF-53-1-90
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR CHEMICAL PROCESSING. Quarterly Report [for] August 10–November 10, [1952]. Chemical Technology Div. D. E. Ferguson. Jan. 7, 1953. Decl. Feb. 16, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

In a study of the removal of Pu from a Pu-producing reactor fuel solution, hexavalent Pu is reduced and precipitated in 24 hrs at 250°C. No difficulties from solvent decomposition would be encountered in the tributyl phosphate solvent in 15 days. However, the formation of surface-active agents and unsaturated compounds will cause some difficulties in solvent recovery. (F.S.)

13640 CF-53-3-208
Oak Ridge National Lab., Tenn.
THOREX PROCESS: CONTINUOUS DISSOLVER DEVELOPMENT PROGRAM QUARTERLY REPORT FOR PERIOD NOVEMBER 1, 1952–FEBRUARY 1, 1953. D. L. Foster. Mar. 23, 1953. Decl. Feb. 14, 1957. 11p. $1.80(ph OTS); $1.80(mf OTS).

13641 CF-53-8-219
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR CHEMICAL PROCESSING QUARTERLY. D. E. Ferguson. Aug. 12, 1953. Decl. Feb. 13, 1957. 19p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

As now envisioned, fuel processing of a two-region, aqueous homogeneous breeder would consist of removal of

rare earths and insoluble fission products from the fuel solution by an inorganic adsorbent. A more complete clean-up of the fuel solution would be accomplished by periodically sending a portion of the fuel through the blanket processing plant. The blanket processing plant would use the Thorex process to isolate U and Th for return to the reactor. This combination of processes will give an economically feasible method of both keeping the neutron losses low and the fuel solution chemically clean. Of the adsorbents tried at both Vitro and ORNL, calcium fluoride still appears the most promising. The solubility of calcium sulfate, a product of the metathesis, in fuel solution was found to be only 0.13 mg/ml at 250°C. Since this solubility is appreciably greater at lower temperatures, an adsorption process using calcium fluoride must be operated near 250°C to avoid precipitation of calcium sulfate inside the reactor. The testing of this adsorption process on a continuous basis at 250°C has not yet been successfully carried out due to U precipitation. This precipitation of U apparently is not due to the presence of calcium fluoride, but rather hydrolysis and reduction in the stainless steel equipment. A material suitable for packing a calcium fluoride adsorption column has been prepared by both fusion of reagent grade calcium fluoride in an inert atmosphere and a cold, hydrostatic pressing procedure. Based on an optimized scheme of blanket processing by Thorex and fuel processing by both calcium fluoride adsorption and in the blanket processing plant, the total cost of processing a two-region, U^{233} breeder is estimated to be 1.3 ml per kwhr, for three percent poisons in the core. For five percent poisons the cost will be less than one ml per kwhr. The estimated cost of processing a one-region reactor is about 30% less on the same basis. (auth)

13642 CF-54-6-169
Oak Ridge National Lab., Tenn.
PROBLEM PROPOSAL FOR HOMOGENEOUS REACTOR PROCESSING. R. E. Leuze and R. A. McNees. June 17, 1954. Decl. Apr. 12, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Problems to be solved and man power needed to develop a chemical flowsheet for K-49 blanket processing are outlined. (F.S.)

13643 CF-54-6-185
Oak Ridge National Lab., Tenn.
D_2O RECOVERY FROM HOMOGENEOUS REACTOR BLANKET AND CORE SOLUTIONS. William L. Carter. June 22, 1954. Decl. Feb. 15. 1957. 6p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A "liquid hot plate" technique has been suggested for use in recovering D_2O from homogeneous reactor blanket and core solutions. The steps in this process are briefly outlined, and, a flow diagram of the process is given. (B.J.H.)

13644 CF-55-3-190(Del.)
Oak Ridge National Lab., Tenn.
UNIT OPERATIONS STATUS REPORT [FOR] MARCH 1955. W. K. Eister. Decl. with deletions Feb. 14, 1957. 56p. Contract W-7405-eng-26. $9.30(ph OTS); $3.60(mf OTS)

Ore Processing. The results from two runs in the 6-in. Higgins contactor using mill pulp from Utex ore are given and discussed. Excer Process. Fabrication of a moving bed reactor and a run in the 4-in. reduction-hydrofluorination reactor are reported. Hope Project. Pulse pumping tests and air lift pumping tests are described, and evaluations of the ORNL hypodermic needle type sampler and the Unibolt disconnect are made. A study of the permeability of a 0.5-in. thick asphalt membrane to acid waste, neutralized waste, and water is presented. Metallex Process. Some test runs using the PDA exchange system are reported. Monex Process. A solvent extraction run is described. Homogeneous Reactor Fuel Reprocessing. The operation of A and B loops and installation and testing of equipment are reported. Hydroclone development and corrosion testing of stainless steel hydroclones are described. (T.R.H.)

13645 CF-56-1-18
Oak Ridge National Lab., Tenn.
ESTIMATION OF ACTIVITY IN HIGH LEVEL PUREX FEEDS. E. D. Arnold. Jan. 4, 1956. Decl. Mar. 13, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Fission product activity levels and solvent damage rates for high level Purex solutions resulting from processing fast reactor fuels are estimated. (C.W.H.)

13646 CF-56-2-81
Oak Ridge National Lab., Tenn.
REMOVAL OF IODINE FROM HOMOGENEOUS REACTORS. D. E. Ferguson. Feb. 16, 1956. Decl. Mar. 16, 1957. 29p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Two suggested methods look both feasible and attractive for the removal of iodine from the fuel of a two-region reactor. The first consists of stripping I from the fuel with O_2 and D_2O vapor, scrubbing iodine from the gas with about 4 gpm D_2O, and finally letting down this small stream of D_2O for removal and storage of the I. For a 5 ft core operating at 360 Mw, this method involves circulating a calculated 275 gpm of fuel solution countercurrent to 4.2 lb moles/min of O_2-D_2O gas to maintain a Xe^{135} poison level of 0.5%. The second method consists of a radiolytic gas let-down system similar to that designed for the HRT, which would take advantage of the stripping of both I and the rare gases by allowing the radiolytic gas to form. This method would require a gas let-down rate of 2600 ml/sec with a liquid let-down rate of about 10 gpm to maintain the Xe^{135} poison level of the TBR at 0.5%. (auth)

13647 CF-56-2-104(Del.)
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR WEEK ENDING FEBRUARY 10, 1956. R. E. Blanco and D. E. Ferguson. Feb. 20, 1956. Decl. with deletions Mar. 15, 1957. 15p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

13648 CF-56-3-33
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR WEEK ENDING FEBRUARY 24, 1956. R. E. Blanco and D. E. Ferguson. Mar. 2, 1956. Decl. Mar. 14, 1957. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

13649 CF-56-3-162
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR WEEK ENDING MARCH 16, 1956. R. E. Blanco and D. E. Ferguson. March 26, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

13650 CF-56-4-41
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR WEEK ENDING MARCH 30, 1956. R. E. Blanco and D. E. Ferguson. Apr. 6, 1956. Decl. Mar. 14, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

13651 CF-56-4-100
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR WEEK ENDING APRIL 6, 1956. R. E. Blanco and D. E.

Ferguson. Apr. 12, 1956. Decl. Mar. 26, 1957. 18p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

13652 CF-56-4-164(Del.)
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR WEEK ENDING APRIL 13, 1956. R. E. Blanco and D. E. Ferguson. Apr. 23, 1956. Decl. with deletions Mar. 14, 1957. 13p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40(mf OTS).

13653 CF-56-6-73(Del.)
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR WEEK ENDING JUNE 1, 1956. R. E. Blanco and D. E. Ferguson. June 6, 1956. Decl. with deletions Mar. 12, 1957. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Progress is reported on the development of Hermex and Metallex Processes, Thorex pilot plant operation, heterogeneous reactor fuel reprocessing, uranium tetrafluoride production by Excer Process, rare earth separation by ion exchange, homogeneous reactor fuel and blanket studies, and protactinium–uranium–thorium processing. (C.W.H.)

13654 CF-56-6-154
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR WEEK ENDING JUNE 15, 1956. R. E. Blanco and D. E. Ferguson. June 27, 1956. Decl. Mar. 12, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

13655 CF-56-6-159
Oak Ridge National Lab., Tenn.
RECOVERY OF URANIUM FROM RECYCLE WASTES BY FLUORIDE VOLATILITY. G. I. Cathers and D. E. Ferguson. June 29, 1956. Decl. Mar. 19, 1957. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The present method of recovering uranium from the scrap obtained in the production of reactor fuel elements consists of obtaining a nitrate solution by dissolving or leaching, solvent extraction, and, finally, precipitation. An alternate method that has been investigated for scrap processing is fluoride volatility. It was found that reaction of MgF_2 slag with elemental fluorine at 400°C volatilized 99% of the contained uranium, provided that good gas-solid contact was obtained. Based on these laboratory results, a process which consists of grinding, screening, fluorinating with fluorine gas at 400°C in an agitated reactor, and collecting the product UF_6 in a cold trap appears to be an attractive method for treating the scrap from the UF_6 and UF_4 reduction steps. The oxide and metal scrap and HNO_3 pickling liquor would require chemical conversion to UF_4 before fluorination, and for this reason the volatility method does not appear to be greatly simpler than the present aqueous method for this type of scrap. (auth)

13656 CF-56-6-174
Oak Ridge National Lab., Tenn.
CHEMICAL DEVELOPMENT STATUS REPORT FOR WEEK ENDING JUNE 22, 1956. R. E. Blanco and D. E. Ferguson. June 22, 1956. Decl. Mar. 12, 1957. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

13657 CF-56-8-20
Oak Ridge National Lab., Tenn.
HRP-BP: HRT UO_2SO_4 BLANKET SOLIDS DISSOLVING AND CORROSION TESTS IN PHOSPHORIC ACID SOLUTIONS. William L. Carter. Aug. 3, 1956. Decl. Mar. 19, 1957. 5p. Contract [W-7405-eng-26.] $1.80(ph OTS); $1.80(mf OTS).

Information is given covering the dissolution of Homogeneous Reactor Test UO_2SO_4 blanket fission and corrosion product solids in concentrated phosphoric and pyrophosphoric acids. (T.W.S.)

13658 CF-56-8-34
Oak Ridge National Lab., Tenn.
STATUS REPORT FOR CHEMICAL DEVELOPMENT, SECTIONS A AND B [FOR] WEEK ENDING JULY 27, 1956. R. E. Blanco and D. E. Ferguson. Aug. 6, 1956. Decl. Mar. 12, 1957. 18p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

13659 CF-56-9-41(Del.)
Oak Ridge National Lab., Tenn.
STATUS REPORT FOR CHEMICAL DEVELOPMENT, SECTIONS A AND B: WEEK ENDING AUGUST 31, 1956. R. E. Blanco and D. E. Ferguson. Sept. 10, 1956. Decl. with deletions Mar. 13, 1957. 10p. $1.80(ph OTS); $1.80(mf OTS).

13660 CF-56-9-113(Del.)
Oak Ridge National Lab., Tenn.
STATUS REPORT FOR CHEMICAL DEVELOPMENT—SECTIONS A AND B: WEEK ENDING SEPTEMBER 14, 1956. R. E. Blanco and D. E. Ferguson. Sept. 26, 1956. Decl. with deletions Mar. 13, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

13661 CF-56-10-58(Del.)
Oak Ridge National Lab., Tenn.
STATUS REPORT FOR CHEMICAL DEVELOPMENT, SECTIONS A AND B: WEEK ENDING SEPTEMBER 28, 1956. R. E. Blanco and D. E. Ferguson. Oct. 8, 1956. Decl. with deletions Mar. 13, 1957. 17p. $3.30(ph OTS); $2.40 (mf OTS).

13662 CF-56-11-80
Oak Ridge National Lab., Tenn.
STATUS REPORT FOR CHEMICAL DEVELOPMENT SECTIONS A AND B: WEEK ENDING NOVEMBER 9, 1956. R. E. Blanco and D. E. Ferguson. Nov. 14, 1956. Decl. Mar. 12, 1957. 7p. Contract [W-7405-eng-26.] $1.80 (ph OTS); $1.80(mf OTS).

13663 CN-1848
Clinton Labs., Oak Ridge, Tenn.
APPLICATION OF DECONTAMINATION STUDIES TO THE DESIGN OF IMPROVED STEPS IN THE COMPLETE $BiPO_4$ PROCESS. Final Report [on] Problem Assignment No. 203-X51C. D. R. Miller, comp. July 15, 1944. Decl. Feb. 16, 1957. 20p. (A-2672). $3.30(ph OTS); $2.40(mf OTS).

Various methods of improving the decontamination in the $BiPO_4$ process have been investigated. The following procedures have been considered: LaF_3, $BaSO_4-LaF_3$, CeF_3, CeF_3-BaSO_4, $BaSO_4$, Ce(IV) and Zr(IV) phosphates, and $BaZrF_6$ scavengers; and complexing of Zr and Cb with Hf in the $BiPO_4$ product precipitation. Increased decontamination by a factor of about 10 at the end of the first cycle was obtained through the use of $BaSO_4-LaF_3$ scavengers. CeF_3, $BaSO_4-CeF_3$, and $BaSO_4$ scavenging gave inconsistent results. Small improvements (about two-fold) were obtained after the first cycle through the use of Ce(IV) and Zr(IV) phosphate scavengers. The use of $BaZrF_6$ in conjunction with LaF_3 and $BaSO_4$ gave no better decontamination than that obtained with $BaSO_4-LaF_3$. Complexing of Zr and Cb with HF in the first cycle product precipitation led to somewhat higher factors for that step, but lower over-all factors at the end of the crossover

cycle. The best decontamination was obtained merely by precipitation of a LaF_3 by-product in the crossover cycle. Other single procedures or combinations gave poorer, or at best comparable, results. Use of the LaF_3 by-product in the crossover cycle led to over-all gamma decontamination factors (at the end of the crossover cycle) of about 2×10^5. (auth)

13664 HW-17519
[Hanford Works, Richland, Wash.].
"25" CRITICALITY CONSULTATION WITH J. W. MORFITT, APRIL 3, 1950. E. T. Merrill and G. Sege. Apr. 4, 1950. Decl. Feb. 19, 1957. 5p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

The application of the critical mass of U^{235} experimental studies was discussed with reference to the application of control in the 25 Process. (F.S.)

13665 HW-18404(Rev.)
Hanford Works, Richland, Wash.
PROPERTIES OF NEUTRALIZED AND CONCENTRATED AQUEOUS WASTE — LOW ACID FLOWSHEETS TBP-HW-NO. 4 AND NO. 5. M. H. Curtis, M. K. Harmon, and V. R. Cooper. July 18, 1950. Decl. Mar. 14, 1957. 13p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

In order to determine the maximum reduction in waste volumes that may be realized after neutralization of RAW, an experimental determination of freezing point versus degree of concentration is required. The ability of the plant to handle and transfer the wastes of the desired degree of concentration requires information on such physical properties as pH, density, viscosity and heat capacity. (auth)

13666 HW-28405
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
THE USE OF PHOSPHATE TO SUPPRESS EXTRACTION OF NIOBIUM AND ZIRCONIUM IN A MODIFIED PUREX PROCESS SYSTEM. Report of Invention. Eugene E. Voiland. June 16, 1953. Decl. Apr. 4, 1957. 2p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

13667 HW-34502(Rev.)
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
INVESTIGATION OF SOLVENT DEGRADATION PRODUCTS IN RECYCLED URANIUM RECOVERY PLANT SOLVENT. R. H. Moore. Apr. 28, 1955. Decl. Feb. 27, 1957. 19p. Contract W-31-109-eng-52. $0.30(OTS).

Flowsheets are presented for the separation and identification of degradation products found in the recycled plant solvent (20% TBP in Shell diluent). It is shown that the impurities arising from the diluent are the principal cause of deterioration in solvent quality. (C.W.H.)

13668 IDO-14224
American Cyanamid Co. Atomic Energy Div., Idaho Falls, Idaho.
REMOVAL OF HEXONE FROM AQUEOUS RAFFINATES. Startup Report No. 42. R. B. Lemon. Feb. 18, 1953. Decl. Mar. 16, 1957. 12p. Contract AT-(10-1)-177. $3.30(ph OTS); $2.40(mf OTS).

The determination of the amount of evaporation required to recover essentially all of the hexone from aqueous raffinate solutions and the amount of hexone remaining in the raffinate solutions at the end of the concentration step was made. (auth)

13669 ISC-298(Rev.)
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT IN ENGINEERING FOR JULY, AUGUST AND SEPTEMBER 1952. Dec. 1, 1952. 12p. Contract W-7405-eng-82. $3.30(ph OTS); $2.40(mf OTS).

13670 KAPL-99
Knolls Atomic Power Lab., Schenectady, N. Y.
OPERATION OF THE "JUNE 9th" TYPE MIXER-SETTLER. Chemical Engineering Laboratory Report No. 3. B. V. Coplan and J. K. Davidson. Nov. 5, 1948. Decl. Feb. 28, 1957. 16p. Contract W-31-109-eng-52. $3.30 (ph OTS); $2.40(mf OTS).

The operability of the six-stage box-type mixer-settler for the Redox Process was demonstrated. Stage efficiencies of 86.6 and 60% were obtained for 1A and 1C Redox bank runs, respectively. The uranium recovery was 99.4%. (C.W.H.)

13671 KAPL-328
Knolls Atomic Power Lab., Schenectady, N. Y.
THE DETERMINATION OF PLUTONIUM IN URANIUM RECOVERED BY REDOX. Walter S. Andrus and David J. Fisher. Apr. 11, 1950. Decl. Apr. 4, 1957. 9p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

A modified lanthanum fluoride procedure is described for the determination of total Pu in Redox stream 1EU. This technique is applicable to analyses of ± 5 parts Pu per 10^9 parts U. (C.W.H.)

13672 KAPL-M-DJS-2
Knolls Atomic Power Lab., Schenectady, N. Y.
CENTRIFUGATION TESTS [ON] KAPL THOREX HEAD-END. Donald J. Smith. Mar. 30, 1955. Decl. Apr. 5, 1957. 15p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40 (mf OTS).

A series of experiments testing the centrifuge properties of MnO_2 cake formed during the KAPL Thorex head-end is described. The effect of varying the acid concentration from 0.25M to 1.00M at 0.8M $Th(NO_3)_4$ and the effect of varying the Th concentration from 0.5M to 1.20M at 0.60M HNO_3 was measured. Centrifugation is satisfactory at greater than 0.5M HNO_3 and less than 1.00M $Th(NO_3)_4$. (auth)

13673 KAPL-M-LLG-8
[Knolls Atomic Power Lab., Schenectady, N. Y.]
HEALTH PHYSICS ASPECTS FOR SPRU OPERATION AT LEVELS OF ACTIVITY GREATER THAN FIVE PER CENT. L. L. German. Nov. 8, 1950. Decl. Feb. 12, 1957. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

13674 KLX-1037
Kellex Corp., New York.
REPORT OF THE INSTRUMENT AND MECHANICAL DEVELOPMENT DEPARTMENT ON FLOW MEASUREMENT AND CONTROL. Job 15. June 30, 1949. Decl. Mar. 1, 1957. 43p. [For Hanford Works]. Contract W-31-109-Eng-52, Subcontract G-148. $16.80(ph OTS); $5.70(mf OTS).

Instrumentation relating primarily to flow-rate measurements in the Redox Process is developed. Emphasis is given to the design and testing of rotameter, caloric, and magnetic induction flowmeters. (K.S.)

13675 KLX-1605
Vitro Corp. of America, New York.
HOMOGENEOUS REACTOR PROCESSING PROGRESS REPORT [FOR] SEPTEMBER 1952. Job 57. Oct. 10, 1952. Decl. May 13, 1957. 39p. Contract AT(11-1)-217. $6.30(ph OTS); $3.00(mf OTS).

A method for recovering D_2O solvent from spent homogeneous reactor UO_2SO_4 fuel solution was developed. A study was made for determining the feasibility and economics of ion exchange separation of fission products from UO_2SO_4 solutions for comparison with alternate decontamination

processes. The development of the most economical process for large scale preparation of a satisfactory homogeneous reactor fuel by conversion of pure UF_6 to UO_2SO_4 dissolved in D_2O is studied. (J.E.D.)

13676 KLX-1606
Vitro Corp. of America, New York.
HOMOGENEOUS REACTOR PROCESSING PROGRESS REPORT [FOR] PERIOD OCTOBER 1–31, 1952. Job 57. Nov. 10, 1952. Decl. May 13, 1957. 41p. Contract AT(11-1)-217. $7.80(ph OTS); $3.30(mf OTS).

13677 KLX-1611
Vitro Corp. of America, New York.
PROJECT SUMMARY D_2O RECOVERY (57-B). Period [Covered]: April 1, 1952–January 31, 1953. Job 57. Apr. 1, 1953. Decl. Feb. 26, 1957. 68p. Contract AT(11-1)-217. $0.45(OTS).

Several proposed alternate methods for recovery of D_2O from a UO_2SO_4 homogeneous reactor fuel solution have been investigated and compared economically for normal to slightly enriched uranium (1% U^{235}). Of the alternate methods, all of which were tested on a laboratory scale, the most efficient and economically attractive consists of evaporating and drying UO_2SO_4 solution in pans to a final temperature of 500°C. Less attractive alternates were drum drying, ion exchange demineralization, and evaporation over a high boiling immiscible liquid. (W.L.H.)

13678 MCW-162
Mallinckrodt Chemical Works, St. Louis.
STUDIES ON THE USE OF CALCINED CX [CRUDE CALCIUM URANATE] AS A PLANT 6 [URANIUM ORE REFINING] FEED. Carl W. Kuhlman, Jr. and C. F. Ritchie. Dec. 31, 1948. Decl. Mar. 27, 1957. 14p. $3.30(ph OTS); $2.40(mf OTS).

Calcination of 22% (U content) crude calcium uranate (CX) at 1000°C resulted in partial sintering of the material with some liberation of SO_3 and in increased bulk density. It was apparently effective in dead burning the gypsum, but it definitely rendered some of the U non-extractable. Calcining at 800°C gave similar results but to a smaller extent. HNO_3 consumptions (hot digestions) averaged 1.46 lb actual HNO_3 per lb U for the material calcined at 1000°C. Digestion filtration filtrates from the 22% CX were treated with $BaCO_3$ for sulfate removal. Sulfate-free filtrates were adjusted and extracted with ether with satisfactory results. Fairly strong acids were found to leach the U rendered "insoluble" by calcining from the cakes. The dissolved U was then recovered by neutralization with caustic soda, but the recovered "soda cake" was low quality, large quantities of acid were needed, and solutions supersaturated with gypsum were encountered. (W.L.H.)

13679 MCW-221
Mallinckrodt Chemical Works, St. Louis.
PILOT PLANT STUDY OF A PROCESS FOR INTRODUCING CX [CALCIUM URANATE FEED MATERIAL] INTO CLARIFIED FEinc FILTRATE. Walter Weber and J. H. Yeager. Sept. 20, 1949. Decl. Mar. 27, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

The proposed process consists of addition of calcium uranate feed (CX) to a quantity of HNO_3 sufficient to (1) solubilize all the U, (2) provide an excess in the filtrate of 0.3 lb HNO_3/gal, and (3) be equivalent to the SO_4 content of the CX. A quantity of clarified FEinc filtrate (okay on Ra) equal to half the volume of HNO_3 is used to thin the CX slurry so that it can be mixed and pumped properly. This slurry is added to FEinc filtrate in such quantity that about 20% of the total U being processed is provided by the CX. The remainder of the process is similar to plant procedures. (W.L.H.)

13680 NAA-SR-1529
Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] JULY–SEPTEMBER 1955. A. G. Buyers, S. J. Yosim, and E. E. Motta, eds. June 1, 1956. Decl. Mar. 1, 1957. 26p. $0.35(OTS).

Decontamination experiments on kg-quantities of long-cooled U fuel were continued. Molten-fluoride salt-scrubbing experiments indicated that extraction of Pu from 60-kg melts would require 1 to 2 hours. Techniques were developed to separate the oxide coatings from oxide-slagged U metal. In small-scale separations studies, U–Pu alloys have been prepared by Ca reduction of CaF_2–LiF–UF_4 systems containing Pu. In other batch experiments in which Pu was extracted from molten U by molten Mg, the distribution coefficient varied when different batches of Mg were used. The causes for the variation are being investigated. In the Th processing studies, the iodide refining experiments on Th–U alloy have shown that Pa and Zr are co-deposited with the Th, while Sr and Ru were not. Solid slagging of the Th–U alloy with ThS_2 and salt scrubbing with $CaCl_2$ or ThF_4 removed Ce, rare earths, Cs, and Sr, leaving Te, Zr, and Ru with the Th–U alloy. The direct evaporation of fission products from Th–U alloy was found to remove Sr, Cs, and Te very well, Ce and rare earths fairly well, and Ru, Zr, and Pa not at all. The evaporation of fission products from a Th–U alloy during arc melting is being studied. (For preceding period see NAA-SR-1515.) (auth)

13681 NYO-5071
Mallinckrodt Chemical Works, St. Louis.
SUMMARY OF EXPERIMENTS ON PHOSPHATE REMOVAL (FROM ORE REFINERY FEEDS). John W. Barnes. Dec. 21, 1944. Decl. Mar. 28, 1957. 5p. Contract W-7405-eng-1. $1.80(ph OTS); $1.80(mf OTS).

The results of 5 experiments on the phosphate removal from ore refinery feeds are briefly summarized. Experimental procedures are not included. (J.E.D.)

13682 ORNL-2210(Del.)
Oak Ridge National Lab., Tenn.
POWER REACTOR FUEL PROCESSING STATUS REPORT FOR OCTOBER 1956. R. E. Blanco, W. K. Eister, and D. E. Ferguson. Jan. 23, 1957. Decl. with deletions Mar. 14, 1957. 20p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Fused Salt–Fluoride Volatility Process. A decontamination factor of $>10^6$ was achieved with a U loss of 0.14%. Studies on embrittlement of Ni are reported. Hermex Process. Dissolution studies of STR II prototype specimens in fused salt systems over Hg containing a $ZnCl_2$–KCl–NaCl matrix are reported. Crystal bar Zr and the STR II specimens dissolved rapidly and smoothly in $AlCl_3$–NaCl containing NH_4Cl or $HgCl_2$. Zircex Process. Hydrochlorination of U–Zr gave good U recovery, but not for U–Nb or U–Al. Data for design of larger equipment are reported. Darex Process. Uranium does not form volatile chloride complexes; the chloride may be distilled off in 3 stages. Tantalum withstands Darex dissolution conditions. General Processing Studies. The dissolution of Zr and Zircaloy-2 in aqueous nitrate-chloride and radiation effects on Amsco were studied. (T.R.H.)

13683 SOD-17
Standard Oil Development Co., Elizabeth, N. J.
PROCESS DESIGN SPECIFICATIONS FOR SOD TYPE MIXER-SETTLER CONTACTING UNITS FOR APPLICATION TO REDOX RECOVERY PROCESS. Dec. 31, 1948. Decl. Mar. 5, 1957. 33p. Contract AT-30-3-GEN-3. $6.30(ph OTS); $3.00(mf OTS).

TRANSURANIC ELEMENTS AND COMPOUNDS

13684 CF-50-11-121

[Oak Ridge National Lab., Tenn.]

A METHOD FOR THE PRELIMINARY ANALYSIS OF PERCOLATION DATA. S. H. Jury. Nov. 20, 1950. Decl. Feb. 13, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A mathematical analysis is presented of the PuF_2 produced from continuous flow of F_2 gas through a small Ni flow reactor containing a small slug of Pu–U. The off-gas was passed through an alumina bed kept at a temperature of 100°C and examined sectionally after each run. (F.S.)

13685 CN-2215

[Clinton Lab., Oak Ridge, Tenn.]

INVESTIGATIONS OF THE CHEMISTRY OF PLUTONIUM. STUDIES ON REACTIONS OF Pu IN UNH SOLUTIONS. Progress Report [on] P. A. No. 213-X40C. J. R. Dam, A. H. Germany, J. Halperin, J. P. Hunt, D. R. Miller, and G. E. Moore. June 30, 1945. Decl. Apr. 5, 1957. 24p. Contract W-7405-eng-39. $4.80(ph OTS); $2.70(mf OTS).

Since the presence of Pu(VI) in the dissolved metal solution would lead to large Pu losses in the $BiPO_4$ extraction, investigations were performed to determine the likelihood of its presence at that step in plant operation. It was found that no Pu(VI) would exist during or immediately after the metal dissolution step, but that under certain conditions following metal dissolution and before extraction, Pu(IV) could become oxidized to Pu(VI). A number of reagents were tested for their efficiency in reducing Pu(VI) to Pu(IV) in uranyl nitrate solutions. HCOOH, U(IV), N_2H_4, and $NaNO_2$ were unsatisfactory for one or more reasons; a Mn(II)-catalyzed $H_2C_2O_4$ reduction was found to be the most satisfactory procedure. (auth)

13686 WKNL-7-26

Kidde (Walter) Nuclear Labs., Inc., Garden City, N. Y.

CRITICAL RADII OF SPHERES AND INFINITE CYLINDERS OF PuF_4 AND PuF_6. Nunzio Tralli. Dec. 22, 1953. Decl. Apr. 10, 1957. 1p. $1.80(ph OTS); $1.80(mf OTS).

TRITIUM AND TRITIUM COMPOUNDS

13687 MonC-35

Clinton Labs., Oak Ridge, Tenn.

THE PRODUCTION AND PURIFICATION OF TRITIUM. Problem Assignment CX1-8. C. V. Cannon, E. Shapiro, G. Jenks, J. Boyle, L. T. McClinton, N. Elliott, C. J. Borkowski, H. S. Pomerance, and R. P. Metcalf. Nov. 29, 1945. Decl. Feb. 28, 1957. 36p. Contract W-35-058-eng-71. $6.30(ph OTS); $3.00(mf OTS).

The production of tritium by the neutron bombardment of LiF is described. The tritium product was purified by passage through a Pd valve. (C.W.H.)

URANIUM AND URANIUM COMPOUNDS

13688 A-1047

Mallinckrodt Chemical Works, St. Louis.

BROWN OXIDE BY-PRODUCT RECOVERY. II. J. V. Opie, W. A. Oppold, and G. C. Reid. Aug. 23, 1943. Decl. Feb. 12, 1957. 20p. $0.30(OTS).

The recovery of uranium (U_3O_8) from residues containing vanadates, molybdates, and phosphates as uranyl salts is described. Recoveries of 92 to 98% were obtained by a process combining caustic leaching, calcination, and nitric acid dissolution. (C.W.H.)

13689 A-2576

Du Pont de Nemours (E. I.) & Co. Jackson Lab., Wilmington, Del.

CHAMBERS WORKS PROCESS FOR THE MANUFACTURE OF PERURANIC ACID. G. M. Richardson. Nov. 15, 1944. Decl. Mar. 18, 1957. 7p. (BP-13). $1.80(ph OTS); $1.80 (mf OTS).

Directions are given for precipitating peruranic acid by means of hydrogen peroxide from uranyl sulfate solutions which have been clarified by filtration. (F.S.)

13690 BC-28

California. Univ., Berkeley. Radiation Lab.

THE SUBLIMATION OF URANIUM TETRACHLORIDE. Paper No. 16. Manfred E. Mueller. [1946?]. Decl. Apr. 8, 1957. 84p. $16.80(ph OTS); $5.70(mf OTS).

A description of experimental studies is presented for the behavior of granular UCl_4 during sublimation under a variety of geometric and thermal conditions. It has been shown that UCl_4 sublimes primarily from the top layer, even when impure, and that the rate of sublimation depends on surface conditions and impurities. The use and effects of applied radiant energy were studied. (W.L.H.)

13691 CC-2092

[Chicago. Univ.] Metallurgical Lab.

PROPERTIES OF URANYL FLUORIDE[1]. Terminal Report–Problem Assignment No. 182 MLC 2601. G. R. Dean. Sept. 11, 1944. Decl. Feb. 16, 1957. 12p. Contract [W-7401-eng-37]. $3.30(ph OTS); $2.40(mf OTS).

Pure UO_2F_2 is made by reaction between anhydrous HF and pure UO_3. Although it is difficult to prepare free of water, it is not very hygroscopic. Prolonged heating at 100 to 105° releases 99.9% of the water in an aqueous solution; the rate of loss of water is very small after saturation is reached. X-ray diffraction data give a density of 6.2 for the solid. The density of a 1.004M solution is 1.274; the density of a saturated solution (5.180M) is 2.400. The densities of D_2O solutions have been calculated from the densities of ordinary aqueous solutions, and the values are compared graphically. Data on the solubility that there is a small increase in the range 25 to 100°. Recovery of pure D_2O from a solution is hampered by hydrolysis of UO_2F_2 to give HF. The HF may be held back by addition of UO_3 or CaO; the use of the latter, however, makes more difficult the recovery of all the D_2O. Molar conductivities of UO_2F_2 solutions at 25°, as well as the specific conductivities of a 1 molar solution containing various concentrations of UO_3, have been determined. The addition of UO_3 lowers the conductivity. The low solubility of UO_4 in UO_2F_2 suggests the possibility of precipitation of that compound during operation of a homogeneous pile containing UO_2F_2 solutions. Data presented indicate that a low concentration of HF is sufficient to prevent such precipitation. The active barium separation method used as a monitor of neutron bombardment of UNH may be applied without modification to UO_2F_2. (auth)

13692 CF-56-5-114

Oak Ridge National Lab., Tenn.

A STUDY OF DILUTE SOLUTIONS IN THE SYSTEM, UO_2SO_4–$CuSO_4$–$NiSO_4$–H_2SO_4–H_2O. F. E. Clark, J. S. Gill, R. Slusher, and C. H. Secoy. May 17, 1956. Decl. Mar. 13, 1957. 6p. Contract [W-7405-eng-26.] $1.80(ph OTS); $1.80(mf OTS).

Studies carried out on the system UO_2SO_4 (0.04m)–$CuSO_4$ (0 to 0.04m)–$CuSO_4$ (0 to 0.04m)–$NiSO_4$ (0 to 0.04m)–H_2SO_4 (0.01 to 0.03m)–H_2O indicated that solid or liquid phases separated at temperatures ranging from 200 to 400°C. (C.W.H.)

13693 FMPC-339
National Lead Co. of Ohio, Cincinnati.
SPRAY COATING OF GRAPHITE MOLDS. George M. Shera. Oct. 19, 1953. Decl. Mar. 13, 1957. 19p. Contract AT(30-1)-1156. $0.30(OTS).

A preliminary report is presented of an improved process for coating graphite molds used in U casting. Equipment, formula, coating preparation, mold preparation, and the coating procedure used are described. (auth)

13694 HW-32410
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
THE IDENTIFICATION AND CHARACTERIZATION OF EXPLOSIVE RESIDUES PRODUCED BY DISSOLUTION OF URANIUM–ZIRCONIUM ALLOYS. W. W. Schulz, F. A. Scott, and E. E. Voiland. Nov. 8, 1954. Decl. Mar. 27, 1957. 36p. Contract W-31-109-Eng-52. $0.40(OTS).

The properties of residues produced by nitric acid dissolution of U–Zr alloys containing 1.5, 4.9, and 8.7 wt. % zirconium were investigated. Residues from annealed alloys consisted principally of a finely divided explosive component which was identified as an intermetallic compound (epsilon phase) with the approximate composition UZr_2. Also present were small amounts of Zr carbide formed in the alloy preparation and an inert material believed to be a solid solution of U and Zr oxides produced by the attack of nitric acid on U–Zr solid solution phases. Partially annealed alloys produced residues with less epsilon content and more inert oxide, while the residues from gamma quenched alloys were non-explosive and consisted primarily of the inert oxide. On the basis of x-ray-diffraction data, the epsilon phase has been assigned a body centered cubic structure with a = 6.16 A. Both desensitization and dissolution occur when the intermetallic compound reacts with nitric acid. Kinetic studies indicate the rate of desensitization to be first order with respect to the explosive compound and nitric acid. The rate of dissolution is markedly slower than the rate of desensitization. Metallographic and x-ray-diffraction data are included. (auth)

13695 K-145
Carbide and Carbon Chemicals Corp., Oak Ridge, Tenn.
PYROHYDROLYSIS IN THE GRAVIMETRIC DETERMINATION OF URANIUM IN URANIUM HEXAFLUORIDE. W. D. Cline and R. H. Lafferty. Mar. 2, 1948. Decl. Mar. 12, 1957. 11p. $3.30(ph OTS); $2.40(mf OTS).

A method for the quantitative conversion of uranyl fluoride to urano-uranic oxide by the use of superheated steam is described. The procedure was designed for the elimination of the time-consuming step of conversion to nitrate used in the analysis of UF_6. A diagram of the comparatively simple apparatus is given, with a quality control chart of routine analysis for both methods. The developed procedure is not only less time consuming but also gives a significantly higher precision. (auth)

13696 K-587
Carbide and Carbon Chemicals Div., K-25 Plant, Oak Ridge, Tenn.
THE ELECTROLYTIC DISSOLUTION OF METALLIC URANIUM. J. A. McLaren, W. D. Cline, H. S. Clinton, J. J. Finley, J. H. Goode, and J. A. Westbrook. Apr. 17, 1950. Decl. Apr. 10, 1957. 11p. Contract W-7405-Eng-26. $1.80(ph OTS); $1.80(mf OTS).

Experiments have shown that U metal can be dissolved in a wide variety of acids, salts, and alkalies by anodic oxidation in an electrolytic cell. Uranium concentrations of 50 to 80 g/l have been obtained with sodium bicarbonate, tartaric acid, or nitric acid as electrolytes. (auth)

13697 K-1114
Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.
THE INTERMOLECULAR INTERACTION OF UF_6 MOLECULES. W. C. DeMarcus and M. P. Starnes. June 23, 1954. Decl. June 19, 1957. 23p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

An expression for the intermolecular potential between two UF_6 molecules is derived. The accuracy of the expression is contingent on the reliability of the available values for the physical properties of the gas and may subsequently have to be adjusted to take into account more reliable information. (L.T.W.)

13698 KLX-1612
Vitro Corp. of America, New York.
PROJECT SUMMARY CONVERSION OF UF_6 TO HOMOGENEOUS REACTOR FUEL (57-D). Period [Covered:] April 1, 1952–January 31, 1953. Job 57. Apr. 1, 1953. Decl. Feb. 26, 1957. 81p. Contract AT(11-1)-217. $0.50 (OTS).

Methods were investigated for the conversion of UF_6 to UO_2SO_4 suitable for use as homogeneous reactor fuel. Throughputs of both 2,000 lb and 11 lb of U per day were considered. The process developed and demonstrated on a laboratory scale for 2,000 lb of U per day throughput of slightly enriched UF_6 (approximately 1% U^{235}) comprises hydrolyzing UF_6 in 70% H_2SO_4, air sparging of the hydrolysis product at 90°C to remove HF, filtering the resultant UO_2SO_4 crystals and calcining the crystals at 500°C. The process developed and demonstrated on a laboratory scale for 11 lb per day throughput of highly enriched UF_6 (over 90% isotopic U^{235}) purity comprises hydrolyzing UF_6 in water, adding a slight excess of H_2SO_4 and drying and calcining at 500°C. (auth)

13699 M-5381
[National Bureau of Standards, Washington, D. C.]
USE OF METAL MOLDS FOR CASTING URANIUM. H. E. Cleaves. [1946?]. Decl. Mar. 12, 1957. 10p. $3.30 (ph OTS); $2.40(mf OTS).

13700 NAA-SR-36
North American Aviation, Inc., Downey, Calif.
STUDIES IN THE IMPREGNATION OF GRAPHITE WITH URANIUM. Gordon N. Steele. June 23, 1950. Decl. Mar. 26, 1957. 26p. Contract AT-11-1-GEN-8. $0.35(OTS).

Experiments are described on the development of homogeneous graphite–uranium bodies, intended for eventual use as reactor fuel elements. Impregnation of graphite with an aqueous solution of uranyl nitrate, with further processing to yield U oxide dispersed in the graphite matrix, has been satisfactorily achieved for 4-in. thick sections. Uniformly of U distribution is improved by precipitating ammonium diuranate, $(NH_4)_2U_2O_7$, within the graphite through treatment with NH_3 gas. Such a process has been used to date for concentrations in the range 1 to 5 mg of U (as oxide) per cm^3 of graphite. Dense, fine-grained graphites of the AGOT type were used. Electron micrographs of powdered, impregnated graphite indicate an oxide particle-size distribution with a maximum at 0.3 to 0.4×10^{-4} cm (0.3 to 0.4 μ). (auth)

13701 NYO-1466
Harshaw Chemical Co., Cleveland.
THE REACTIVITY OF URANIUM OXIDES DECEMBER 31, 1951. Karl S. Willson. Feb. 15, 1952. Decl. Feb. 27, 1957. 22p. Contract W-7405-eng-276. $0.35(OTS).

The occasional poor conversions in the reduction of UO_3 to UO_2 and the hydrofluorination of UO_2 to UF_4 was studied. Results of experiments indicate that this variable chemical

activity may be due in part to different crystalline forms of UO_3 as a result of variations in the denitrating operation. (M.H.R.)

13702 NYO-3796
Metal Hydrides Inc., Beverly, Mass.
REDUCTION OF URANIUM OXIDES. QUARTERLY REPORT NO. 4. Sidney Johnson. Oct. 19, 1953. Decl. Feb. 27, 1957. 14p. Contract AT(30-1)-1350. $0.25(OTS).

Means of reduction of uranium oxides to uranium hydride powder were further studied. On the basis of previous experience, emphasis was upon the reduction of UO_2 by CaH_2, which not only has given some of the best results, but also is considered to be the reaction most readily adaptable to a continuous process. Three methods of evaluation which are considered the most promising are being pursued. These are: (1) a U assay method whereby the UH_3 powder is dissolved directly in a measured excess of $CeSO_4$; (2) the density of U metal prepared by hot-pressing UH_3 powder; and (3) a direct determination of oxygen by displacing this element with BrF_3. In addition, the A. E. C. New Brunswick Laboratories agreed to do a limited number of oxygen determinations by the vacuum fusion technique on U metal produced in this laboratory. It is certain that one or more of these methods will be suitable for a complete evaluation of UH_3 powder. (auth)

13703 NYO-5055(Del.)
Mallinckrodt Chemical Works, St. Louis.
GRAPHITE LINER PROGRAM. Report No. 3. J. V. Opie. July 15, 1944. Decl. with deletions Feb. 26, 1957. 14p. $3.30(ph OTS); $2.40(mf OTS).

Equipment and procedures are described which were employed in the experimental casting of uranium in graphite crucibles. (C.H.)

13704 ORNL-2043
Oak Ridge National Lab., Tenn.
DETERMINATION OF TRIVALENT URANIUM IN FLUORIDE SALT MIXTURES BY THE MODIFIED HYDROGEN EVOLUTION METHOD. J. C. White, A. S. Meyer, Jr., W. F. Vaughan, W. J. Ross, and D. L. Manning. Feb. 28, 1956. Decl. Mar. 1, 1957. 21p. Contract W-7405-eng-26. $0.50(OTS).

13705 WAPD-PMM-404(Del.)
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
MINUTES OF URANIUM OXIDE PANEL MEETING. J. Glatter and R. C. Koch. Oct. 18, 1955. Decl. with deletions Mar. 5, 1957. 9p. $1.80(ph OTS); $1.80(mf OTS).

Irradiation tests on UO_2 and UO_2–ceramic systems are reported and discussed. (T.R.H.)

13706 Y-388
Carbide and Carbon Chemicals Corp. Y-12 Plant, Oak Ridge, Tenn.
THE CONVERSION OF URANYL AMMONIUM PHOSPHATE TO URANIUM HEXAFLUORIDE IN A VERTICAL VAPOR PHASE REACTOR. Research and Development Division Progress Report for February and March 1949. C. C. Haws and W. M. Leaders. Apr. 18, 1949. Decl. Mar. 6, 1957. 12p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

13707 Y-876(Del.)
Carbide and Carbon Chemicals Co. Y-12 Plant, Oak Ridge, Tenn.
DEVELOPMENT OF HIGH DENSITY URANIUM DIOXIDE POWDERS. Harvey T. Kite and David W. Smith. May 28, 1952. Decl. with deletions May 7, 1957. 17p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Various procedures for the preparation of UO_2 powders are reviewed. Blending of various sieve sizes of UO_2 and coprecipitation of U with magnesia are also reported. (L.T.W.)

WASTE DISPOSAL

13708 KLX-44
Kellex Corp., New York.
THE FEASIBILITY OF CONTINUOUS CHELATION. Job 11. Apr. 29, 1949. Decl. Mar. 28, 1957. 51p. Contract AT-30-1-Gen-169. $9.30(ph OTS); $3.60(mf OTS).

A survey of the feasibility of the use of chelation for continuous processing of (a) dissolver solution, (b) solutions resulting from Redox Process wastes, and (c) solutions from the treatment of wastes from the present process for plutonium recovery and decontamination, super clean-up of uranium, zirconium recovery and packaging, uranium recovery and decontamination, and fission product recovery has been made. New experimental work on batch and continuous extraction in a horizontal countercurrent flow extractor is reported and previous work is reviewed. The conclusion is drawn that continuous chelation processes are probably feasible for all of the above. Possible schemes for incorporation of chelation into process design are outlined and discussed. The chief advantage of these schemes lies in the possibility of reducing or completely eliminating the bulk of contaminated salts in waste solutions which present serious problems in handling or storage. In this connection, consideration is given to the self-salting properties of uranium nitrate in the hexone–water–nitric acid system. Some aspects of the significance of specific solvent effects upon the extractibilities of metallic ions as chelates are discussed. (L.M.T.)

ENGINEERING

HEAT TRANSFER AND FLUID FLOW

13709 CF-52-3-20
Oak Ridge National Lab., Tenn.
BUBBLE BEHAVIOR WITHIN LIQUIDS FLOWING IN TUBES CONTAINING 'U' BENDS. H. F. Poppendiek. Mar. 3, 1952. Decl. Aug. 14, 1957. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

13710 CF-54-8-225
Oak Ridge National Lab., Tenn.
SOME OBSERVATIONS MADE ON CAVITATING SODIUM FLOW IN A VENTURI. J. M. Trummel. Aug. 31, 1954. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Cavitating sodium flow was obtained in the venturi of an isothermal loop at temperatures from 1215 to 1475°F. Cavitation was detected in three ways: by observing the minimum venturi throat pressure, by observing the loop pressure drop, and by listening to the flow noise. Cavitation occurred promptly when venturi throat pressure was reduced to within about 1.5 psia of vapor pressure. Venturi wall damage due to cavitation has not been determined. (auth)

13711 KAPL-M-AJH-3
Knolls Atomic Power Lab., Schenectady, N. Y.
METALLURGICAL EXAMINATION OF GRISCOM RUSSELL BENT-TUBE TWO-TUBE STEAM GENERATOR NO. 1. W. L. Fleischmann and A. J. Herd. Feb. 3, 1956. 6p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The stability of the heat transfer bond and the ability of the tube attachments to withstand sudden temperature changes in the bent-tube-two-tube steam generator were tested by thermal cycling. Galling signs in the inner surfaces of the outer tubes where they were rolled into the tube sheets were found. This condition to a less extent was also found where the inner tubes were rolled into their tube sheets, but as this condition was expected it does not reflect on the quality of the steam generator. (F.S.)

13712 KAPL-M-DDA-2
Knolls Atomic Power Lab., Schenectady, N. Y.
TEMPERATURE TRANSIENTS IN EVAPORATORS. D. D. Adams. [1956]. 10p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

A simplified method for evaluating temperature transients in evaporators is presented. The method is based on the thermal velocity and effective heat capacity concepts. The differential equation for the transient temperature distribution is developed, and one method of solution is indicated. (auth)

13713 KAPL-M-DRM-12
Knolls Atomic Power Lab., Schenectady, N. Y.
THERMO-ELASTIC STABILITY ANALYSIS. D. R. Miller. Apr. 12, 1956. 6p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

If a heat producing plate in a cooling channel is not centered so as to achieve symmetry of temperatures, the temperature distribution through the plate will be asymmetrical. This asymmetry initiates a tendency to develop curvature which would increase the asymmetry; giving rise to a possibility of thermo-elastic instability. Equations are developed for describing this condition. (D.E.B.)

13714 KAPL-M-EDL-61
Knolls Atomic Power Lab., Schenectady, N. Y.
FREE CONVECTION FLOW STUDIES. R. G. Jacoby, G. C. Capelianis, and J. J. Stewart. July 1, 1954. 23p. Contract [W-31-109-eng-52]. $4.80(ph OTS); $2.70(mf OTS).

A study was conducted in order to (1) discover whether free convection currents would occur in a narrow cylindrical annulus, (2) observe the pattern of such convection currents, and (3) study the effects of peripheral restrictions at various levels on the convection currents. (L.T.W.)

13715 KAPL-M-HB-33
Knolls Atomic Power Lab., Schenectady, N. Y.
NOTES OF THE BETA EXPERIMENT. H. Brooks. May 25, 1950. Decl. Feb. 27, 1957. 7p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS)

Numerical calculations concerning the heat transfer characteristics of reactor fuel elements are briefly reviewed. (B.J.H.)

13716 KAPL-M-JJK-1
Knolls Atomic Power Lab., Schenectady, N. Y.
SODIUM VAPOR TRANSPORT IN A CLOSED SYSTEM. J. J. Kauzlarich. June 17, 1953. Decl. Mar. 12, 1957. 11p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80 (mf OTS).

The transport of Na in a closed system is studied, and a Labyrinth-type vapor seal is investigated for possibilities of limiting Na vapor transport. Theory predicts that such a seal will be ineffective because it will collect Na vapor, the large cross section does not limit diffusion or evaporation, and the seal will not offer resistance to pressure surges. (F.S.)

13717 KAPL-M-LRB-3
Knolls Atomic Power Lab., Schenectady, N. Y.
DETECTION OF NUCLEATE BOILING BY FLUX VARIATION MEASUREMENT. L. R. Boyd. July 23, 1956. 13p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Nucleate boiling in a reactor neutron flux can be detected by means of an ion chamber located in the vicinity of the boiling. The variation of the moderator density by the formation of vapor bubbles superimposes a characteristic fluctuation on the local neutron flux, which is reflected in the signal from the ion chamber. (auth)

13718 KAPL-M-RES-29
Knolls Atomic Power Lab., Schenectady, N. Y.
LUMPED METAL HEAT CAPACITY. J. P. Fraser. July 16, 1956. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Calculation of thermal transients in the primary or secondary reactor systems must take into account the thermal heat capacity of metal changing temperature with the water. A relatively simple basis for deciding on the representation of associated metal heat capacity in analog or digital computer calculations is described. (L.T.W.)

13719 NAA-SR-47
North American Aviation, Inc., Los Angeles.
TRANSIENT HEAT TRANSFER IN HETEROGENEOUS REACTORS. W. C. Parrish. Dec. 19, 1949. Decl. Feb. 27, 1957. 31p. Contract AT-11-1-gen-8. $0.35(OTS).

An investigation is made of the transient heat generation and transfer in a thermal heterogeneous reactor, cooled and moderated by D_2O, after super-criticality produced by a sudden release of 2% excess k. Two cases are considered: In Case I, the D_2O is at a mean temperature of 170°F during steady state operation. In Case II, the steady state D_2O temperature was that of saturation (212°F). The analysis shows that because of the critical point on the curve of film coefficient versus temperature differential, for heat transfer to a liquid with boiling, the fuel elements of Case I will overheat and fail quite rapidly. Although the rate of heat transfer to the D_2O is relatively large, such heat is utilized in increasing the sensible heat of the moderator with little change in moderator density. Consequently, the increase in reactor period by the time of fuel element failure is insignificant. In Case II, all heat transferred to the moderator is available for phase change of the latter with appreciable decrease in density. According, the rate of increase of reactor period is sufficient to prevent the differential temperature from approaching the critical point. Consequently, the fuel element of Case II is safe from overheating (i.e., assuming that subsequent relatively long-time residual heat generation is removed at the rate generated). (auth)

13720 NDA-31(Del.)
Nuclear Development Corp. of America, White Plains, N. Y.
REACTOR HEAT TRANSFER PROGRESS. [Issue] No. 12. (15J-17). John E. Viscardi, ed. Oct. 30, 1956. Decl. with deletions May 8, 1957. 24p. Contract AT(30-1)-862. $4.80(ph OTS); $2.70(mf OTS).

Abstracts of classified papers to be presented at the Reactor Heat Transfer Conference are given, and a correlation study of subcooled and quality burnout data for water in forced circulation through tubes and ducts is included. (J.H.M.)

13721 WAPD-PWR-CP-2351(Del.)
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
IN-PILE TEST SECTION PRESSURE DROP SUMMARY. [Oct. 1956]. Decl. with deletions Mar. 12, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

Pressure drop characteristics were plotted for the WAPD-CR-5 loop in NRX Reactor and WAPD-13 and -15 loops in the Materials Testing Reactor. (D.E.B.)

13722 WAPD-RM-169

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

CYCLIC VARIATION OF REACTOR POWER AS A MEANS OF DETERMINING FILM COEFFICIENT VARIATION. G. M. Anderson. Feb. 1953. Decl. Mar. 13, 1957. 20p. Contract AT-11-1-GEN-14. $3.30(ph OTS); $2.40(mf OTS).

A method of sinusoidal oscillation of reactor power as a means of determining film coefficient variation is described. The results show the expected dependence of outlet water temperature and neutron level on film coefficient. For the case considered, a 100% change in film coefficient produced a 20% change in the amplitude of the water temperature oscillation at one cycle per second. At half power, the outlet temperature excursion is 4°F for a 10% oscillation of reactor power. The corresponding variation with film coefficient is 1°F. Neutron levels are insensitive to variations in film coefficient and are therefore of little use as an indication of such variations. Transient excitation of the reactor in lieu of sinusoidal excitation is briefly considered. (auth)

MATERIALS TESTING

13723 ANL-4900

Argonne National Lab., Lemont, Ill.

A CAPSULE DESIGN FOR EXPERIMENTAL HIGH-FLUX IRRADIATIONS OF FUEL MATERIALS. J. Howard Kittel and P. Tedeschi. Oct. 6, 1952. Decl. July 15, 1957. 8p. Contract W-31-109-eng-38. $1.80(ph OTS); $1.80(mf OTS).

A Zr capsule for experimental irradiation of fuel materials is described which contains, for heat removal, enough NaK alloy to cover the specimen. The capsule exhibits negligible corrosion by water and NaK, low cross section for thermal neutrons, adequate thermal conductivity, and easy leak inspection after final closure. (J.E.D.)

13724 HW-25330(Del.)

Hanford Works, Richland, Wash.

BEHAVIOR OF S. I. R. FUELS IRRADIATED IN SODIUM. (SPECIAL REQUEST KAPL-79: THE "BETA EXPERIMENT"). Final Report [on] Production Test 105-180-P. J. B. Lambert. Aug. 12, 1952. Decl. with deletions Apr. 1957. 119p. Contract W-31-109-Eng-52. $18.30(ph OTS); $6.00(mf OTS).

Irradiations of fuel materials and coil springs for the Submarine Intermediate Reactor were performed in dry process tubes in a Hanford pile. The fuel specimens and controls were in capsules of Na, and temperature of the fuel was maintained at 450°C when feasible. The coil springs were in He gas at 350°C, and were under compression during irradiation. Eleven of the capsules contained fuel in the form of enriched U metal or oxide. Five control capsules contained natural or depleted U metal. Nine metallic fuel specimens were in direct contact with the Na, whereas eight of the metallic or oxide fuel specimens were sealed in tubes of Zr, Ti, Fe, or type 347 stainless steel which were immersed in Na. Total irradiations varied from 1.3×10^{19} to 4×10^{20} n/cm^2. Handling of the materials at Hanford and the conditions of irradiation are discussed. Data are tabulated on performance of heaters and thermocouples. No data from post-irradiation investigations of the materials are included. (C.H.)

13725 IDO-16344

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

RADIATION DAMAGE TO CERTAIN WATER RESISTANT PAINTS USED AS PROTECTIVE COATINGS IN GAMMA FACILITIES. W. C. Francis. Jan. 18, 1956. 9p. Contract AT(10-1)-205. (PTR-36). $1.80(ph OTS); $1.80(mf OTS).

Four potential underwater protective coatings have been subjected to radiation in the MTR Gamma Facility. One sample was discarded because of poor adhesion. The three remaining samples showed evidence of deterioration but were still considered satisfactory after an exposure of 5×10^7 to 1.0×10^8 roentgens. After exposures of 1.0 to 3.7×10^9 roentgens, two samples were badly blistered and the third powdery. At this stage all three samples were considered unsatisfactory for further use as a protective coating. (auth)

13726 KAPL-M-RBM-2

Knolls Atomic Power Lab., Schenectady, N. Y.

VELOCITY SHOCK TRANSMISSION IN TWO DEGREE SERIES MECHANICAL SYSTEMS. Robert B. McCalley, Jr. Feb. 1, 1956. 20p. Contract W-31-109-Eng-52. $6.30(ph OTS); $3.00(mf OTS).

Previous work by Mindlin and Crede on shock transmission through isolators assumed that the mass of the element was negligible in comparison to the mass of the chassis upon which it was mounted. A more exact analysis of the problem is given without any restriction on the relative masses of the element and the chassis. Contour map charts for both shock transmissibility through the isolator and relative magnification of the chassis deflection have been prepared with a range of 10^6 in both spring and mass ratios for an undamped system so that changes required for improved shock performance are readily determined. The charts show that only the stiffness ratio should be changed in some cases while, in others, both the mass and stiffness ratios should be altered. This new analysis shows that the resonance condition can be tolerated in many instances without the necessity for damping predicted by the approximate analysis. For example, with the element mass as small as 1% of the chassis mass, the shock magnification of an undamped system is only 10, whereas the earlier study for a negligible element mass predicted a magnification of 17 even with 1% critical damping on both element and chassis. (auth)

13727 KAPL-M-WMC-1

Knolls Atomic Power Lab., Schenectady, N. Y.

LABORATORY SIMULATION OF THERMAL EFFECTS IN FUEL ELEMENTS. W. M. Cashin. Feb. 11, 1954. Decl. Mar. 12, 1957. 13p. Contract W-31-109-Eng-52. $3.30 (ph OTS); $2.40(mf OTS).

Work is described which simulates the thermal experience of a fuel element and thereby permits prediction of fuel element behavior without resorting to pile experiments. (L.T.W.)

13728 KAPL-M-WVJ-2

Knolls Atomic Power Lab., Schenectady, N. Y.

FRICTION AND WEAR TESTS IN WATER. W. V. Johnston and D. T. Groleau. Oct. 18, 1956. 10p. Contract W-31-109-Eng-52. $1.80(ph OTS): $1.80(mf OTS).

This is a summary of the friction and wear tests made at KAPL to date covering tests made in hot water at temperatures from room temperature to 550°F and sliding speeds of approximately 2 and 20 feet per minute. A spring load of 330 grams was used. Tests have been conducted on 17-4PH heat treated to hardness of Rc35 and Rc43 and type 410 stainless steel of Rc40 and Rb90 hardness. The results indicate that wear on the disk is greatest at temperatures around 300°F where the type of protective surface present in air at room temperature has been worn off and the rate of a metal-water reaction is too slow to provide a protec-

tive layer. Greater wear has been observed on the harder of the two 17-4PH steels when they are tested against one another. (auth)

13729 WAPD-B-1
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
TEST CONDUCTED ON OPEN TYPE SHIM ROD MAGNET. B. Levy. May 5, 1950. Decl. Mar. 18, 1957. 8p. $1.80 (ph OTS); $1.80(mf OTS).

In determining the feasibility of using the open-type electromagnet as a shim rod magnet, tests indicated that the use of this type magnet is not feasible with the insulation now available. (F.S.)

13730 WAPD-CP-1121
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
PROPOSAL FOR DYNAMIC IN-PILE CYCLIC FUEL MATERIALS TESTS AT MTR. E. S. Lembersky. May 10, 1955. Decl. Mar. 12, 1957. 12p. $3.30(ph OTS); $2.40 (mf OTS).

WASTE DISPOSAL

13731 CF-55-8-76(Del.)
Oak Ridge National Lab., Tenn.
HEAT TRANSFER IN WASTE BASINS. S. H. Jury. Aug. 11, 1955. Decl. with deletions Mar. 11, 1957. 16p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Estimates are made of the heat transfer in proposed out-of-door radioactive waste basins. Calculations are outlined, and complete numerical results are included in appendices. (B.J.H.)

13732 WASH-4 & Suppl.(Del.)
Parsons (Ralph M.) Co., Pasadena, Calif.
REPORT OF INVESTIGATIVE STUDIES AND TENTATIVE DESIGN CRITERIA FOR DESTRUCTOR PLANT. T. W. Burtt and W. E. Grebe. Dec. 8, 1949. Decl. with deletions Mar. 4, 1957. 55p. Contract AT(29-1)-540. $9.30 (ph OTS); $3.60(mf OTS).

Details of a proposed "destructor plant" for the disposal of contaminated wastes at Los Alamos are presented. Design criteria, feed flow, engineering features, safety provisions and cost estimates are included. Appended details cover air-decontamination methods and systems. (T.W.S.)

MINERALOGY, METALLURGY, AND CERAMICS

13733 KAPL-M-CCR-1
Knolls Atomic Power Lab., Schenectady, N. Y.
REMOTE BRAZE SEAL STUDY. C. C. Ripley and E. C. Reynolds. Aug. 3, 1950. 4p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Two 2-in. stainless steel pipes were used to demonstrate a satisfactory brazed joint technique. (D.E.B.)

CERAMICS AND REFRACTORIES

13734 BMI-763
Battelle Memorial Inst., Columbus, Ohio.
BERYLLIA AND BERYLLIA–BERYLLIUM BODIES FOR MODERATING CORES IN FUEL ELEMENTS. R. E. Long, W. H. Duckworth, H. A. Saller, J. T. Stacy, R. A. Miller, and H. Z. Schofield. July 31, 1952. Decl. Apr. 2, 1957. 16p. Contract W-7405-eng-92. $0.25(OTS).

Beryllia bodies were chemically compatible with Mo and urania-bearing Mo, on the basis of visual examination and chemical analyses of the test specimens. In tests of physical compatibility, painted-on coatings of Mo powder, sintered in place on the cylindrical surface of prefired beryllia discs, withstood thermal-cycling tests by maintaining contact with the underbody. Shrunk-on rings, machined from Mo, stretched and loosened from the beryllia discs during these tests. Dense structures of beryllia-Be were not obtained in the range of compositions with more than 25% BeO. Some densification of bodies containing less than 25% BeO was obtained, but the best bodies, from the viewpoint of high density, were in the range of 5% or less BeO. These had little of the desirable rigidity at high temperatures imparted by refractory BeO. (auth)

13735 CF-52-1-60(Del.)
Oak Ridge National Lab., Tenn.
SELECTION OF ORGANIC COATING FOR CELL WALLS IN THE IDAHO CHEMICAL PROCESSING PLANT. F. L. Culler. Jan. 7, 1952. Decl. with deletions Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Plastics were studied and recommendations made for cell coatings of sufficient radiation stability and adequate decontamination potential to be used in a cell operating at a level of 3000 g of U^{235} per day. (F. S.)

13736 CT-1914
[Chicago. Univ. Metallurgical Lab.]
HEAT TRANSFER CALCULATIONS FOR DIE CASTING AL–SI COATINGS ON "W" SLUGS. Report on a Phase of Problem 154-ML-52-1. G. M. Brown. Jan. 29, 1945. Decl. Feb. 20, 1957. 16p. Contract W-7401-eng-37. $3.30(ph OTS); $2.40(mf OTS).

Die casting Al–Si on U was investigated as an alternate procedure for preparing Hanford elements. Preliminary calculations were made to estimate suitable operating conditions for die casting and to serve as a basis for interpreting the results. The operation consists of forcing molten Al–Si through an annular space between a slug and a steel shell and allowing it to freeze. A reservoir is provided at the end of the annulus for excess Al–Si. Calculations were based on a film thickness of 2.0 mils and a linear velocity of 166 ft per sec. (D.E.B.)

13737 KAPL-1475
Knolls Atomic Power Lab., Schenectady, N. Y.
SILICA–URANIUM FUEL SYSTEMS. W. M. Cashin and H. G. Sowman. Feb. 15, 1956. Decl. Mar. 28, 1952. 18p. Contract W-31-109-Eng-52. $0.20(OTS).

Uranium oxide–SiO_2 systems containing 4 to 10 wt.% uranium were prepared by fusion of SiO_2 with powdered UO_2, and by impregnation of Corning porous VYCOR with uranyl nitrate with subsequent firing to give U_3O_8 in dense, fused silica. Irradiation of unenriched material produced no gross changes. Samples containing U^{235} showed cracking upon irradiation in lead–bismuth alloy, but did not release fission products. (auth)

13738 LA-1133(Del.)
Los Alamos Scientific Lab., N. Mex.
SLIP CASTING PROCESS FOR MAGNESIA CRUCIBLES. A. G. Allison and M. R. Nadler. June 15, 1950. Decl. with deletions Feb. 28, 1957. 15p. Contract W-7405-Eng-36. $3.30(ph OTS); $2.40(mf OTS).

The production of drain cast and solid cast MgO refractories is described in detail. Specifications for raw materials, procedures for preparation and casting of the slip, drying and burning of the cast refractories are given. (auth)

13739 WAPD-126

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

THE FABRICATION OF DENSE URANIUM DIOXIDE CYLINDRICAL COMPACTS BY COLD PRESSING AND SINTERING. J. Glatter, H. R. Hoge, and B. E. Schaner. Oct. 14, 1955. Decl. Mar. 12, 1957. 23p. Contract AT-11-1-GEN-14. $0.35(OTS).

Pressing and sintering characteristics of Mallinckrodt hydrogen-reduced UO_2 powder are presented. Punch and die materials and factors influencing punch and die design are described. Density and shrinkage data for sintered UO_2 are given. A graphite resistor sintering furnace is described. (auth)

CORROSION

13740 BMI-1045

Battelle Memorial Inst., Columbus, Ohio.

CORROSION IN 650 F DEGASSED WATER OF URANIUM–MOLYBDENUM ALLOYS CONTAINING IMPURITY ADDITIONS. Edward W. Cawthorne, Warren Berry, James E. Reynolds, Robert I. Jaffee, and Robert S. Peoples. Oct. 6, 1955. Decl. Feb. 13, 1957. 68p. Contract W-7405-eng-92. $10.80(ph OTS); $3.90(mf OTS).

A marked increase in corrosion resistance was found in the U–10.5 wt.% Mo alloys with ternary additions of 0.05 to 0.10 wt.% Si, 0.5 to 1.0 wt.% Zr, 0.25 to 1.0 wt.% Sn. Corrosion rates were as low as –0.01 mg/(cm²)(hr), and the corrosion life was approximately 110 to 125 days which was double that observed for the most resistant binary (as-melted U–10.5 wt.% Mo). The following ternary additions had little or no effect on corrosion behavior of the alloys studied: 0.01 to 0.025 wt.% Si, 0.001 to 0.25 wt.% Zr, 0.001 to 0.10 wt.% Sn, 1 to 10 ppm H_2, 65 to 360 ppm O_2, 20 to 820 ppm N_2. Maximum corrosion resistance could not be correlated with any given heat treatment, but, in general, aging was found to decrease corrosion resistance drastically when large quantities of gamma transformed to alpha U and epsilon. On the other hand, some of the as-cast alloys with rather large amounts of transformation products showed up well in corrosion tests. The discontinuous failures in corrosion tests are associated with hydrogen contamination during corrosion and its subsequent precipitation as needles or plates. Zirconium in excess of about 0.5 wt.% appears to suppress the hydride precipitation during corrosion of U–10.5 wt.% Mo–0.5 wt.% Zr and U–12 wt.% Mo–0.5 wt.% Zr alloys. In the cast series of alloys, it was observed that the greatest improvement in corrosion resistance could be associated with the elements (Si, Sn, and Zr) which decreased the aging response or had no effect upon it. There was no indication that corrosion behavior was sensitive to the U melting stock, the type of crucible in which the biscuit was melted, the quenching rate, or the homogenization treatment. (auth)

13741 BMI-1087

Battelle Memorial Inst., Columbus, Ohio.

THE EFFECTS OF TERNARY ALLOYING ADDITIONS ON THE CORROSION RESISTANCE OF EPSILON-PHASE URANIUM–ZIRCONIUM ALLOYS. James E. Reynolds, Warren E. Berry, Horace R. Ogden, Robert S. Peoples, and Robert I. Jaffee. Apr. 27, 1956. Decl. Mar. 22, 1957. 36p. Contract W-7405-eng-92. $6.30(ph OTS); $3.00(mf OTS).

The corrosion rate in 680 F water of the U–50 wt. % Zr binary alloy was found to be –0.20 mg/(cm²)(hr), and that of the U–40 wt. % Zr binary alloy was –0.34 mg/(cm²)(hr). Both alloys corroded uniformly, with no evidence of discontinuous failure. Normal variations in interstitial content in either alloy had no significant effect on corrosion behavior. Ta additions, in the range of 0.2 to 5 wt. %, improved the corrosion rate of the U–50 wt. % Zr, with a minimum rate of –0.06 mg/(cm²)(hr) for the 5 wt. % Ta alloy. The 5 wt. % Ta addition to the U–40 wt. % Zr alloy decreased the corrosion rate of the base to –0.11 mg/(cm²)(hr) in the epsilonized condition only. In other conditions, the 5 wt. % alloy failed discontinuously. All other additions to both bases either had no effect or decreased corrosion resistance. These included Al, Cr, Fe, Mo, Ni, Pt, Sn, Ti, W, and V additions. (auth)

13742 CF-52-3-207

Oak Ridge National Lab., Tenn.

CORROSION TESTS: SILVER LOADED GRAPHITE, BABBITT LOADED GRAPHITE, II. H. I. Kraig. Mar. 27, 1952. Decl. Apr. 2, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Corrosion tests of silver- and babbitt-loaded graphite in U soup show the babbitt-loaded material more suitable corrosion-wise at high soup concentrations. Either material is suitable at low concentrations such as would be found in seal or bearing installations. Use of either material in direct contact with soups of concentrations greater than 20g U/l is not recommended. (F.S.)

13743 CF-53-1-199

Oak Ridge National Lab., Tenn.

HRP DYNAMIC CORROSION STUDIES. SUMMARY OF RUN J-7. J. C. Griess and R. E. Wacker. Jan. 20, 1953. Decl. Feb. 18, 1957. 3p. $1.80(ph OTS); $1.80 (mf OTS).

13744 CF-53-1-256

Oak Ridge National Lab., Tenn.

RESULTS OF CORROSION TESTS OF TYPE 347 STAINLESS STEEL WELDMENTS IN SOLUTIONS SUPPLIED BY THE CHEMICAL TECHNOLOGY DIVISION. C. C. Vreeland. Jan. 24, 1953. Decl. Apr. 2, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Stainless steel weldments for the Thorex Pilot Plant were corrosion tested in Thorex AW stream and 3-20 decontamination solutions. Uniform weight losses were observed for 3 specimens which were examined metallographically and with a low-power microscope. No excessive attack was noted on the weld metal or base metal. (T.R.H.)

13745 CF-56-2-37

Oak Ridge National Lab., Tenn.

HRP DYNAMIC CORROSION STUDIES. Summary of Run D-61: 3.7m UO_2SO_4, 3.7m Li_2SO_4, 330–350°C. J. C. Griess, R. S. Greeley, and S. R. Buxton. Feb. 9, 1956. Decl. Mar. 26, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Results are reported for dynamic corrosion loop testing of stainless steel, Ti, Zircaloy-2, and several Al, Cr, Sn, and V alloys of Ti in 3.7m UO_2SO_4 + 3.7m Li_2SO_4 + 1000 to 2000 ppm O_2. (T.W.S.)

13746 CRD-T2C-63

California Research and Development Co., Livermore, Calif.

CORROSION PROPERTIES OF ZIRCONIUM AND ZIRCONIUM ALLOYS. M. H. Boyer. Oct. 23, 1951. Decl. Apr. 2, 1957. 8p. (LWS-24170). $1.80(ph OTS); $1.80(mf OTS).

Zirconium is resistant to corrosion in NaK up to 500°C, but it shows high resistance to water corrosion up to 600°F and to superheated steam up to 750°F if it is free of impurities. Zr alloys with 3 to 6% tin show a high resistance

to water up to 600°F and to superheated steam to 750°F and are not seriously affected by the minor amounts of impurities which affect pure Zr. Welding and bonding Zr and Zr–Sn alloys produce corrosion resistant joints if the surfaces are properly prepared and atmospheric gases are excluded during the processes. (F.S.)

13747 CT-3030
[Chicago. Univ. Metallurgical Lab.]
INVESTIGATION OF THE GALVANIC CORROSION OF 304 STAINLESS STEEL, 2S ALUMINUM AND 72S ALUMINUM. Wayne W. Binger. June 2, 1945. Decl. Feb. 16, 1957. 28p. Contract W-7401-eng-37. $4.80(ph OTS); $2.70(mf OTS).

Electrochemical data were obtained in a flowing system on the galvanic corrosion of 304 stainless steel, 2S Al, and 72S Al couples, and these data are correlated with data obtained in stirred beaker tests under similar conditions in simulated river water. (L.T.W.)

13748 HW-14955
Hanford Works, Richland, Wash.
CORROSION INVESTIGATIONS OF REDOX PILOT PLANT EQUIPMENT AT OAK RIDGE NATIONAL LABORATORIES. W. W. Koenig. Nov. 3, 1949. Decl. Feb. 20, 1957. 14p. $3.30(ph OTS); $2.40(mf OTS).

Corrosion encountered in the Oak Ridge pilot plant was small with the exception of the dissolver which showed considerable corrosion of the inside weld and of the welds securing the slug basket to the bottom of the vessel. The slug crash plate and service tubes also showed weld corrosion. (F.S.)

13749 HW-28491
[General Electric Co. Hanford Atomic Products Operation, Richland, Wash.]
STATIC CORROSION TESTS OF FABRICATION MATERIAL FOR D-12 WASTE EVAPORATOR. D. F. Shepard. June 25, 1953. Decl. Feb. 20, 1957. 8p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Three types of stainless steel (309 SCB, Carpenter 20, 304-L) were exposed to both the liquid and vapor phase of each of three types of solutions. The time of exposure totaled ten days during which the specimens were cleaned and weighed periodically. The three synthetic D-12 waste solutions contained respectively chrome-high chloride, chrome-low chloride and high chloride sans chromium. The solutions were renewed at each weighing. Calculation and tabulation of results indicate the order of acceptance of the three metal types tested to be 309-SCB, 304-L and Carpenter 20. However, in the light of physical handling (welding, etc.), economic and other metallurgical factors this order may be altered. (auth)

13750 KAPL-M-GEG-5
Knolls Atomic Power Lab., Schenectady, N. Y.
CORROSION OF T-410 STAINLESS STEEL IN 680°F DEOXYGENATED, DEIONIZED, STATIC WATER. G. E. Galonian and D. C. Belouin. May 4, 1956. 14p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

Type 410 stainless steel has been proposed as a fuel element cladding material for water-cooled reactors. To provide corrosion data to evaluate its resistance in 680°F pure water, preliminary corrosion tests have been made. These consisted of static autoclave tests to determine the steady-state rate of corrosion and resistance to stress and crevice corrosion. Corrosion-time data show that an average steady-state rate of metal loss of about 24 mg/dm^2-mo is reached after 400 hours' exposure. One thousand-hour tests indicate that T-410 is resistant to stress and crevice corrosion. The test results show that T-410 is not as corrosion resistant as the austenitic stainless steels but it does show some promise as a possible cladding material. Further work is required to supplement the corrosion data and confirm the resistance to stress and crevice corrosion. (auth)

13751 KAPL-M-HOS-2
Knolls Atomic Power Lab., Schenectady, N. Y.
CORROSION RESISTANCE AND COATINGS FOR CORROSION PROTECTION OF AISI 410 STEEL. Henry Suss. Jan. 6, 1956. 12p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Corrosion data are tabulated for unclad and clad S3G steel control drive components in dynamic and static coolant water at 300 to 650°F. (T.W.S.)

13752 KAPL-M-JGG-1
Knolls Atomic Power Lab., Schenectady, N. Y.
CORROSION STUDIES OF CRYSTAL BAR ZIRCONIUM IN ORGANIC COOLANTS. J. G. Gratton. Oct. 18, 1956. 10p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80 (mf OTS).

Bench-scale experiments were conducted to investigate the corrosion resistance of crystal bar Zr in diphenyl and monoisopropyldiphenyl with added impurities. Pyrolytic damage to the coolant after 1000 hr at boiling temperatures was also studied. (L.T.W.)

13753 KAPL-M-WWK-8
Knolls Atomic Power Lab., Schenectady, N. Y.
RADIATION-INDUCED CORROSION AND THE POSSIBLE USE OF CORROSION INHIBITORS IN STEAM GENERATING EQUIPMENT. W. W. Kendall and P. H. Klein. Feb. 3, 1956. 10p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

13754 KLX-1040
Kellex Corp., New York.
INSTRUMENT AND MECHANICAL DEVELOPMENT DEPARTMENT REPORT ON DYNAMIC CORROSION TESTING PROGRAM, JOB 15. June 30, 1949. Decl. Feb. 26, 1957. 21p. Contract W-31-109-eng-52, Subcontract G-148. $4.80 (ph OTS); $2.70(mf OTS).

Work is summarized on the problem of dynamic corrosion of process equipment in general, and valves and valve seats in particular, due to the action of the corrosive radioactive solutions which are to be handled in the proposed Redox Process Plants. (F.S.)

13755 WAPD-104
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
THE HIGH TEMPERATURE OXIDATION OF ZIRCALOY IN WATER. W. A. Bostrom. Mar. 19, 1954. Decl. Mar. 21, 1957. 19p. Contract AT-11-1-GEN-14. $0.25(OTS).

The oxidation rates of Zircaloy-2 and Zircaloy-B maintained at temperatures between 1300°C and 1860°C while submerged in water have been measured. The observed rates are approximately those which would be expected from the extrapolation of existing data for Zr in air and the reaction does not become violent or autocatalytic in nature even at temperatures above the melting point. (auth)

13756 WAPD-C-115
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
CRUD MEMORANDUM NO. 14. D[onald] M. Wroughton. Jan. 26, 1953. Decl. Feb. 22, 1957. 10p. $1.80(ph OTS); $1.80(mf OTS).

Additional data substantiating the inhibiting properties of dissolved H_2 are presented by both WAPD and Babcock and Wilcox from loop tests. Cleaning of crud from one Chalk River test loop is discussed. (L.M.T.)

13757 WAPD-FE-85(Rev.)
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
THE EFFECT OF STABILIZED URANIUM FILAMENT SIZE IN MULTIFILS ON THE TYPE OF CORROSION FAILURE. Report of Experiment PWR 11C1. J. Halapatz. June 15, 1954. Decl. Apr. 15, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

The experimental fabrication of compartmented fuel bearing rods by the extrusion of multifil billets is described. Multifil rods were successfully produced with 10% Zr–U fuel located in filaments in a geometric array, separated by Zircaloy-2 cladding. Stock suitable for corrosion testing was prepared. (auth)

13758 WAPD-FE-762
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
SUMMARY OF CORROSION TESTING RESULTS OF DEFECTED CLAD RODS. W. B. Haynes. Apr. 18, 1955. Decl. Mar. 19, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

Results from corrosion tests of defected Zircaloy-2 clad PWR fuel rods in 650°F degassed water are given. (T.R.H.)

13759 WAPD-PWR-CP-1814
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
EROSION OF UO_2 FUEL MATERIAL. L. A. Waldman and J. M. Lojek. Feb. 15, 1956. Decl. Mar. 21, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

Activity release from failed UO_2 elements as a result of erosion by a high-velocity coolant stream was obtained for both as-ground and fractured surfaces of cold-pressed and sintered UO_2. Erosion of exposed surfaces due to clad defects and ruptures does not contribute a significant proportion of activity. (F.S.)

13760 WAPD-RM-116
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
GENERAL CORROSION OF WAPD CRYSTAL BAR ZIRCONIUM. PART I. EFFECT OF TEST CONDITIONS. K. M. Goldman and D. E. Thomas. Mar. 21, 1952. Decl. Mar. 12, 1957. 14p. Contract AT-11-1-Gen-14. $3.30 (ph OTS); $2.40(mf OTS).

Under the conditions normally employed in the corrosion testing of WAPD crystal bar Zr and Mark I strip coupons at 600°F for 7 days, certain anomalous results have arisen. These anomalies were most likely occasioned by lack of control of such variables as surface condition, air in the autoclave, temperature, etc. In an attempt to explain this behavior, a systematic study of various test conditions has been made. The variables investigated include: distance and contact between specimens in degassed and aerated water; local boiling; residual Alconox; N in the testing environment; carbonyl iron, Na_2SO_4, NaF, and Cl contamination in the water; and convection currents. The tests were conducted for 7 days at 600°F. The test time of 7 days was chosen so that these tests would be directly comparable with production tests. Surface condition was kept constant throughout these tests. (auth)

13761 WAPD-RM-160(Rev.)
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
CORROSION IN A LARGE STAINLESS STEEL CIRCULATING WATER SYSTEM AT 475°F. (WAPD-D-LOOP TEST). P. Cohen and V. W. Thompson. Apr. 6, 1953. Decl. Mar. 18, 1957. 56p. Contract AT-11-1-GEN-14. $0.50(OTS).

These tests confirm that forged 347 stainless steel piping corrodes appreciably in H_2O at 475°F, containing small amounts of dissolved O_2 and H_2, forming primarily insoluble corrosion products of which a large part appears as "crud," finely divided magnetite, containing Ni and Cr oxides. At high dissolved H_2 contents, the rate of formation of loose solid corrosion products is markedly decreased, and suspended "crud" concentrations are low. In the absence of radiation and at a crud concentration of 0.25 ppm, deposition on Zr surfaces was essentially zero at 27-fps velocity, and measurable but not significant at a velocity of 6 fps. At low concentrations of crud of the order of 0.06 ppm, deposition was negligible even at 6-fps velocity. For reactor systems, where flux may cause deposition, operation under the conditions of the degassed run of this test would be likely to result in serious fouling of fuel elements. Operation with H_2 appears to be a feasible means of eliminating the crud problem. (auth)

13762 Y-B15-19
[Oak Ridge National Lab., Y-12 Area, Tenn.]
CORROSION TEST BY FLUORIDE SALTS. H. J. Buttram, R. O. Hutchinson, and P. J. Hagelston. Jan. 16, 1951. Decl. Mar. 21, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Corrosion tests have been run on a number of different types of stainless steel, Mo, Ni alloys and Nb in LiF–UF_4 eutectic at 700°C for 420 hours and in NaF–UF_4 eutectic for 600 hr. (W.L.H.)

METALS AND METALLURGY

13763 ANL-4316(Del.)
Argonne National Lab., Lemont, Ill.
METALLURGY DIVISION REPORT FOR JANUARY, FEBRUARY, AND MARCH 1949. Mar. 31, 1949. Decl. with deletions Mar. 11, 1957. 96p. Contract W-31-109-eng-38. $15.30(ph OTS); $5.40(mf OTS).

13764 ANL-5110(Del.)
Argonne National Lab., Lemont, Ill.
DEVELOPMENT OF A PROCESS TO PRODUCE ZIRCONIUM HANFORD TYPE PROCESS TUBING BY ROLL FORMING AND INERT ARC WELDING. R. A. Noland and G. B. O'Keefe. May 1, 1953. Decl. with deletions Feb. 12, 1957. 45p. Contract W-31-109-eng-38. $7.80 (ph OTS); $3.30(mf OTS).

The development of methods for producing Zr Hanford-type process tubing by roll forming and He arc welding flat strip to which appropriate rails had been resistance welded is described. Tubing was made on a Yoder-type tube mill. Dimensional correction was effected by cold swaging on a shaped mandrel. (T.R.H.)

13765 ANL-5339
Argonne National Lab., Lemont, Ill.
HIGH TEMPERATURE STRENGTH ZIRCONIUM AND TITANIUM BASE ALLOYS FOR FUEL ELEMENT JACKETING. Final Report—Metallurgy Program 4.4.6. K. F. Smith and H. H. Chiswik. Feb. 1956. Decl. Apr. 2, 1957. 16p. Contract W-31-109-eng-38. $0.30(OTS).

Titanium and zirconium base-alloys have yield strengths in the temperature range 650 to 800°C equal to or greater than type 347 stainless steel and have excess of 10% room temperature elongation. The Zr-base alloys are not extensively attacked by Na at 660°C but are not resistant to degassed water at 360°C. Titanium-base alloys are corrosion resistant to Na at 660°C and degassed water at 360°C. All the above alloys are weldable by inert arc methods. Thermal conductivity of Ti- and Zr-base alloys range from 0.019 to 0.022 cal-cm/cm^2-sec.°C mean temperature of 130°C, this compares with 0.039 for type 347 stainless steel. (auth)

13766 ANL-5341
Argonne National Lab., Lemont, Ill.
THE CASTING AND FABRICATION OF BINARY ALLOYS OF URANIUM WITH MOLYBDENUM, ALUMINUM, SILICON, TITANIUM, AND VANADIUM. Final Report–Metallurgy Programs 1.1.1 and 1.1.3. Work completed: December 1949. R. E. Macherey and R. J. Dunworth. June 15, 1956. Decl. Apr. 2, 1957. 26p. Contract W-31-109-eng-38. $0.35(OTS).

The melting, casting, and rolling to rods of U–Al, U–Mo, U–Si, U–Ti, and U–V alloys are described. The U–Mo alloys included concentrations of Mo from 0.5 to 6.0 wt. %; all other alloying element additions were below 1 wt. %. Effects of small additions of alloying elements on the growth and surface roughening of U, resulting from thermal cycling, thermal data, and other physical properties are discussed. (R.V.J.)

13767 BM-II-80
Northwest Electrodevelopment Lab., Albany, Oreg.
ZIRCONIUM PROGRESS REPORT FOR THE PERIOD OF SEPTEMBER 15–DECEMBER 15, 1953. Jan 20, 1954. Decl. Feb. 12, 1957. 39p. Contract AT-(11-1)-140. $0.40(OTS).

Production figures on Hf, Zr, and Zr alloys and hot water corrosion data for Zr alloys are reported. Results obtained in the study of derived high-purity concentrates of Zr minerals are summarized. (For preceding period see BM-II-73.) (J.E.D.)

13768 BM-II-90
Northwest Electrodevelopment Lab., Albany, Oreg.
ZIRCONIUM PROGRESS REPORT FOR THE PERIOD OF MARCH 15–JUNE 15, 1954. July 9, 1954. Decl. Feb. 12, 1957. 36p. Contract AT-(11-1)-140. $0.40(OTS).

The production of Hf and Zr and improvement in the yields and reclamation of scrap are reported. Studies were made on the corrosion of Zr and Zr alloys in water at 600 to 1000°F and steam at 750°F and the effects of the addition of metals on the corrosion resistance. A summary of the results in the study of derived high-purity concentrates of Zr minerals is given. (J.E.D.)

13769 BMI-720
Battelle Memorial Inst., Columbus, Ohio.
A STUDY OF THORIUM-BASE ALLOYS. R. M. Goldhoff, H. R. Ogden, and R. I. Jaffee. Dec. 26, 1951. Decl. Apr. 2, 1957. 57p. Contract W-7405-eng-92. $9.30(ph OTS); $3.60(mf OTS).

Dilute alloys of thorium with up to 5 atomic per cent of various additions were investigated. Most additions were ineffective in strengthening thorium. The only really effective ones were carbon and indium. They could double the yield strength in the annealed condition. Cold work is effective in strengthening thorium. The thorium alloys are even more responsive to cold-work strengthening. Fusion welding of thorium, known to be poor, can be improved markedly by dilute additions of molybdenum. Thorium–uranium alloys with up to 50% uranium were also investigated. The entire range of compositions is readily workable, and the larger additions of uranium provide for a substantial increase in strength. (auth)

13770 BMI-721
Battelle Memorial Inst., Columbus, Ohio.
TUNGSTEN-ARC WELDING OF AMES THORIUM AND THORIUM ALLOYS. R. E. Monroe, D. C. Martin, and C. B. Voldrich. Jan. 15, 1952. Decl. Apr. 2, 1957. 38p. Contract W-7405-eng-92. $0.35(OTS).

Arc-welding tests were made on Ames thorium sheet using the inertgas-shielded tungsten-arc-welding process. The Ames thorium was found to be very susceptible to weld cracking. By reducing restraint to a minimum or by applying pressure transverse to the weld during welding, it was possible to make single-pass welds. Multipass welds could not be made. Single-pass weldments had the same strength as unwelded sheet but considerably less elongation and reduction of area. A large number of alloys made with Ames thorium were given a weldability screening test. The results of these tests indicated that a number of alloys had good weldability. Larger scale welding tests were made on an alloy containing 2.2 at. % molybdenum. Single- and multipass tungsten-arc welds could be made in this alloy without weld cracking. The strength properties of weldments were the same as those of the unwelded base plate. (auth)

13771 BMI-752
Battelle Memorial Inst., Columbus, Ohio.
ALLOYS OF URANIUM WITH ZIRCONIUM, CHROMIUM, COLUMBIUM VANADIUM, AND MOLYBDENUM. H. A. Saller and F. A. Rough. June 19, 1952. Decl. Feb. 13, 1957. 41p. Contract W-7405-eng-92. $0.40(OTS).

Information on five uranium binary alloy systems has been surveyed. These systems are the alloys of U with Zr, Cr, Nb, V and Mo. The equilibrium diagrams are discussed, and where available, data have been included on diffusion studies, cladding experiments, and mechanical properties. (auth)

13772 BMI-770(Del.)
Battelle Memorial Inst., Columbus, Ohio.
MECHANICAL PROPERTIES OF ZIRCONIUM AND ZIRCONIUM-URANIUM ALLOYS CONTAINING TIN. A. D. Schwope, G. T. Muehlenkamp, and W. Chubb. Sept. 29, 1952. Decl. with deletions Feb. 17, 1957. 18p. Contract W-7405-eng-92. $3.30(ph OTS); $2.40(mf OTS).

The mechanical properties of a large number of zirconium alloys have been investigated. These alloys include zirconium–tin and zirconium–tin–uranium. Tensile tests have been run on these alloys at room temperature, 500 and 600°F. Hot-hardness data have been obtained from room temperature to 1600°F, with particular emphasis on the hot-rolling range. Creep tests and fatigue tests have been run on a number of zirconium–tin alloys at 500°F. (auth)

13773 BMI-802
Battelle Memorial Inst., Columbus, Ohio.
ELECTROPLATING ON THORIUM. J. G. Beach, W. C. Schickner, L. E. Vaaler, and C. L. Faust. Jan. 9, 1953. Decl. Feb. 13, 1957. 16p. Contract W-7405-eng-92. $0.30(OTS).

The protection of thorium in hot water was studied. Copper electrocladding on thorium offers promising protection against hot water. The corrosion of thorium in 95°C water was reduced appreciably by a thorium phosphate film. The results of this abbreviated investigation offer a basis for the development of a method of protecting thorium at elevated temperatures in corrosive media. (auth)

13774 BMI-805
Battelle Memorial Inst., Columbus, Ohio.
THE FABRICATION OF THORIUM TUBES. H. A. Saller and J. R. Keeler. Jan. 16, 1953. Decl. Apr. 4, 1957. 8p. Contract W-7405-eng-92. $0.25(OTS).

Thorium tubes were cold drawn from extruded tube stock and also from drilled hot-rolled rod. Total reductions up to approximately 85% were accomplished by conventional tube-drawing methods, without an intermediate anneal. The hard-drawn tubes had high tensile strength, but the ductility as indicated by flaring and flattening tests, was low. (auth)

13775 BMI-881
Battelle Memorial Inst., Columbus, Ohio.
BONDING OF URANIUM AND ZIRCONIUM ALLOYS. H. O. McIntire, H. J. Hucek, and G. K. Manning. Oct. 23, 1953. Decl. Apr. 2, 1957 21p. Contract W-7405-eng-92. $0.25 (OTS).

Fabricating conditions for bonding Zr and U alloys by roll cladding were studied. Good bonding was obtained with a minimum reduction of 3 : 1 for cores of unalloyed U, U–20 w/o Zr, or U–50 w/o Zr. As measured by a shear test, the bond was not obviously improved by increasing the amount of reduction to 7 : 1. With unalloyed U cores, the optimum temperature for rolling appeared to be 1175 to 1250°F. Moderately higher temperatures resulted in wavy or cracked cores. U–20 w/o Zr cores cracked when rolled at temperatures of 1175°F or lower. Good results were obtained for temperatures of 1250 to 1450°F. Cores of U–50 w/o Zr did not crack and appeared to bond well at temperatures between 1100 and 1450°F. Temperatures outside this range were not evaluated. The best core configurations resulted with the lower reduction ratios. Also, the U–20 w/o Zr cores gave better results than the unalloyed or U–50 w/o Zr cores. (auth)

13776 BMI-887
Battelle Memorial Inst., Columbus, Ohio.
PROTECTION OF URANIUM: VAPOR-DEPOSITED COATINGS. I. E. Campbell, E. M. Sherwood, C. F. Powell, and R. P. Jones. Nov. 24, 1953. Decl. Feb. 25, 1957. 20p. Contract W-7405-eng-92. $0.30(OTS).

The most satisfactory vapor-deposited coatings on U were obtained by a displacement-diffusion process similar to pack chromizing, employing the vapors of the lower zirconium iodides. Coatings of Zr-U alloy up to 5.9 mils thick were obtained during a 24-hr treatment at 1050°C. These coatings, of undetermined composition, were adherent, nonporous, and exhibited corrosion rates in boiling H_2O of the order of 0.001 to 0.006 mg/cm^2 hr. The average life of the coatings in boiling H_2O is about 500 hr. Cr was deposited on U by thermal decomposition of CrI_2 vapor. Nb was deposited by H_2 reduction of $NbCl_5$ vapor at reduced pressure. Mo was deposited by thermal decomposition of molybdenum carbonyl. Zr was deposited by thermal decomposition of, or displacement from, ZrI_4 vapor by disproportionation of the vapors of lower Zr iodides (ZrI_2 + ZrI_3) and by reduction of the lower Zr iodides with Mg and Zn vapors. This latter group of coatings was unsatisfactory because of porosity, poor adhesion, or the presence of underlying salt deposits. (auth)

13777 BMI-991
Battelle Memorial Inst., Columbus, Ohio.
ALUMINUM CLADDING LONG URANIUM PLATES BY SOLID-STATE BONDING. H. A. Saller, R. F. Dickerson, and R. J. Carlson. Mar. 29, 1955. Decl. Mar. 1, 1957. 20p. Contract W-7405-eng-92. $3.30(ph OTS); $2.40(mf OTS).

Solid-state bonding techniques were investigated as a possible method of obtaining metallurgical bonding between long flat U cores and Al cladding. Because Al-U compounds are brittle, another metal is required between the core and clad to develop a strong, ductile bond. Of the materials surveyed, Ni, electrodeposited on the core, appears to have the most favorable characteristics. Cores measuring 3 by 5 by 0.180 in. were successfully bonded to Al cladding by single-step pressing, and cores measuring 3 by 18 by 0.180 in. were successfully bonded to Al cladding by multiple-step pressing. Maximum pressing conditions of 6000 psi at 950°F were maintained to prevent measurable core deformation. Good dimensional control was obtained for both single-step and multiple-step pressing. No special atmosphere was required during pressing when Ni plate was the bond-facilitating layer. (auth)

13778 BMI-1018
Battelle Memorial Inst., Columbus, Ohio.
VAPOR DEPOSITED ZIRCONIUM ON URANIUM. Carroll F. Powell, Robert P. Jones, Fleet T. Girod, and Ivon E. Campbell. July 20, 1955. Decl. Feb. 14, 1957. 30p. Contract W-7405-Eng-92. $4.80(ph OTS); $2.70(mf OTS).

Low-carbon U was given protective coatings by packing the specimen in Zr and subjecting it to I_2 vapor at 0.5 to 15 mm mercury pressure at 1050°C. The most protective of the coatings was obtained with a 24 hr treatment, giving coatings 3 to 13 mils in thickness. Good coatings were formed only at 1000 C or above. Coating compositions fell in the range $UZr_{0.7-3}$. The protection afforded by these coatings in 100°C water was highly variable, with exposure periods ranging from 200 to 600 hr, averaging 300 hr, before accelerated corrosion occurred. Higher carbon U (0.04 to 0.05 wt.% C) zirconized readily only when prealloyed with 1 wt.% of Zr. Coating periods from 3 to 6 hr gave good coatings only on specimens alloyed with 0.35 at.% Cr or 1 wt.% Zr. Such specimens consistently showed low corrosion rates [0.001 to 0.016 mg/(cm^2)(hr)] over 300 to 500 hr periods. Coated specimens had a coarse grain structure and this was not refined in the few instances where coated specimens were heat treated. The minimum corrosion rate exhibited by pack-zirconized coatings was about 0.0003 mg/(cm^2)(hr). General intergranular penetration of the coating by water vapor appeared to set the maximum low-corrosion period obtainable with a 10-mil-thick pack-zirconized coating at around 2000 to 3000 hr. (auth)

13779 BMI-1043(Del.)
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING SEPTEMBER 1955. Russell W. Dayton and Clyde R. Tipton, Jr. Oct. 1, 1955. Decl. with deletions Feb. 13, 1957. 72p. Contract W-7405-eng-92. $12.30(ph OTS); $4.50(mf OTS).

Progress is reported on the following investigations: effect of compositional variables on corrosion properties of Mo–U alloys; fuel–element end closures; effect of UO_3 pre-treatment on the sinterability of UO_2; effect of S additions to NH_3 on corrosion behavior of stainless steels; hydration of UO_3; preparation and properties of UO_2; corrosion behavior of $MoSi_2$ and $MoSi_2$–30 wt. % UO_2 compacts; thermal fracture of SiC–base materials; mechanical properties and corrosion behavior of U–Zr alloys; fabrication of cermet-base fuel elements; effect of impurity and alloy additions on corrosion properties of Zr and Sn–Zr alloys; internal friction in U; high-temperature stability of Zr alloy hydrides; and oxidation resistance of Nb alloys. (C.W.H.)

13780 BMI-1049
Battelle Memorial Inst., Columbus, Ohio.
RESISTANCE-WELDING METHODS FOR END CAPPING PWR URANIUM OXIDE FUEL RODS. David C. Martin, William H. Kearns, and Harry E. Pattee. Oct. 17, 1955. Decl. Apr. 2, 1957. 24p. Contract W-7405-eng-92. $3.30(ph OTS); $2.40(mf OTS).

A survey of resistance-welding methods for end capping PWR UO_2-cored fuel rods and a brief study of nondestructive testing methods for resistance-welded rods have been made for the Westinghouse Atomic Power Division. Flash, upset, and spot welding were investigated to determine their suitability for making the tube-to-end cap joints. Brief studies were made of welding-schedule variables and joint designs. Simulated fuel rods were fabricated by flash

and upset welding. Physical and mechanical properties of flash- and upset-welded joints were studied by tension tests, hydrostatic tests, hardness tests, corrosion tests, x-ray radiography, and metallographic examinations. Welds made adjacent to UO_2 pellets were studied metallographically and were corrosion tested in 750°F steam. Strong, sound joints were made by both flash and upset welding. Flash welding proved to be more consistent than upset welding. The welding conditions appeared to be less critical. With upset welding, difficulty was experienced in controlling the heat input. Some joints overheated and melted, others did not weld completely. In both types of welding, the end oxide pellet cracked during welding. After 28 days of exposure to 750°F steam, both flash and upset welds showed faint white corrosion products at the weld joints. Spot welding as a means of joining the tube to the end cap did not prove successful. The electrode force required to obtain a weld produced excessive deformation in the tube wall. Shunting of welding current through the preceding weld contributed to inconsistent results. Flash- and upset-welded joints were inspected nondestructively by eddy-current methods. Large separations between the tubes and end caps were the only type of flaws detected. Flash-welded joints produced erroneous indications except when the internal flash was removed. (auth)

13781 BMI-1066
Battelle Memorial Inst., Columbus, Ohio.
A STUDY OF ALUMINUM–URANIUM ALLOYS.
Henry A. Saller, Ronald F. Dickerson, Frank A. Rough, Ellis L. Foster, Arthur A. Bauer, and John R. Lulay. Jan. 19, 1956. Decl. Mar. 25, 1957. 52p. Contract W-7405-eng-92. $7.80(ph OTS); $3.30(mf OTS).

The liquidus for Al alloys containing 13 to 25 wt.% has been redetermined by metallographic examination, thermal analyses, and electrical-resistance measurements of alloys in this range. Metallographic examination of the alloys containing 9 to 19 wt.% indicates the eutectic composition is 13.2 ± 0.2 wt.% U. The composition limit of the peritectic horizontal, obtained by extrapolation of the liquidus curve, is 16.3 wt.% U. The liquidus temperature of a 24.5 wt.% U alloy is 954°C. Eutectic and peritectic temperatures are 640 and 732°C, respectively. In addition to the phase-diagram studies, a method of casting sound, homogeneous, good-surfaced 16 wt.% U alloy sleeve castings was developed. Cooling rates of the alloy in the mold were studied and it was established that a minimum cooling rate of 70°C per min would result in homogeneous castings. The effect of cooling rates was confirmed with the use of a casting especially designed for this purpose. (auth)

13782 BMI-1136(Del.)
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING SEPTEMBER, 1956. Russell W. Dayton and Clyde R. Tipton, Jr. Oct. 1, 1956. Decl. with deletions Mar. 12, 1957. 69p. Contract W-7405-eng-92. $10.80 (ph OTS); $3.90(mf OTS).

Progress is reported in: development of materials for reactors, fuel-element development, PWR development, metallurgy, Zr alloys. (T.R.H.)

13783 BMI-T-7
Battelle Memorial Inst., Columbus, Ohio.
ELECTROPLATING OF METALS ON THORIUM. J. G. Beach, G. R. Shaer, and C. L. Faust. Feb. 1, 1949. Decl. Feb. 14, 1957. 16p. Contract W-7405-eng-92. $0.30(OTS).

A method is described for electroplating adherent metal coatings directly on Th. The principal new development relates to the pretreatment of the Th surface prior to electroplating. This pretreatment consists of anodic pickling in HCl plus chemical pickling in H_2SO_4. Adherent Cu, Zn, Ni, Cr, Ag, Pb, Sn, Au, In, Rh, and Al can be electroplated on the pretreated Th surface by methods described. Iron was also studied, but the deposits obtained were not comparable with the others in quality. (auth)

13784 CE-345
[Chicago. Univ.] Metallurgical Project.
TECHNOLOGICAL DIVISION [AND MIT METALLURGICAL GROUP AND AMES METALLURGICAL AND PRODUCTION GROUP] REPORT FOR MONTH ENDING NOVEMBER 15, 1942. Decl. Apr. 2, 1957. 46p. Contract [W-7401-eng-37]. (A-732). $7.80(ph OTS); $3.30(mf OTS).

13785 CF-53-3-183
Oak Ridge National Lab., Tenn.
FABRICATION OF SPHERICAL PARTICLES. A. Levey. Mar. 19, 1953. Decl. Feb. 14, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Two general methods for the preparation of spherical particles of Cu–U and Ni–U alloys were examined both in this investigation and work previously reported. One general method, that of ejection of molten metal or spraying of molten wire is characterized by the formation of spheres with little oxidation, good retention of U but with wide variation in particle size. The second general method is the momentary melting in an electric arc or molten salt of wire particles precut to the desired volume. The latter methods require considerable refinement to eliminate oxidation, porosity and overheating. (auth)

13786 CF-56-10-18(Del.)
Oak Ridge National Lab., Tenn.
NICKEL–MOLYBDENUM ALLOY NEWSLETTER. Oct. 15, 1956. Decl. with deletions Mar. 1, 1957. 19p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The oxidation rates of Mo–Ni alloys are summarized. The effect of carbon on the tensile ductility of annealed Mo–Ni alloys is shown. Fabrication of Mo–Ni base alloys is described. Data on the tensile properties of various Mo–Ni alloys at 1300°F are tabulated. The production of commercial size heats of Mo–Ni alloys is discussed. Several Mo–Ni test plates were welded, and hardness studies of aged welded joints and mechanical properties of as-welded joints are reported. (L.T.W.)

13787 CT-1501
[Iowa State Coll., Ames.]
METALLURGICAL RESEARCH REPORT FOR PERIOD OF APRIL 10 TO MAY 10, 1944. June 12, 1944. Decl. Feb. 18, 1957. 15p. Contract [W-7405-eng-82]. (A-2084). $3.30(ph OTS); $2.40(mf OTS).

Phase studies in the Mo–U, Cu–U, Mg–U, and Sn–U systems are reported. A hydrogen etching hot stage for metallographic samples is discussed. (L.M.T.)

13788 CT-1697
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS REPORT ON METALLURGY OF TUBALLOY [Uranium]. May 1, 1944. Decl. Mar. 15, 1957. 50p. Contract [W-7405-eng-92]. (A-2427). $0.40(OTS).

Progress is reported on dilatometric studies of extruded and rolled U; the effect of pressure on the $\alpha-\beta$ transformation temperature; studies on the U–nitrogen system; hydrogen content of rolled rod; solubility of carbide in U; procedures for α extrusion of U; determination of hot hardness of α and γ-extruded and cast U metal; the diffusion bonding of Al jackets with Al–Si; the hot pressing of Be powders; the effect of annealing on the tensile properties of U; and the effects of casting conditions on the grain size of 3% Al–Be alloys. (C.H.)

13789 CT-2253
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS REPORT ON METALLURGY OF TUBALLOY [URANIUM] TO UNIVERSITY OF CHICAGO. Oct. 1, 1944. Decl. Feb. 19, 1957. 19p. Contract [W-7405-eng-92]. $4.80(ph OTS); $2.70(mf OTS).

Dimensional stability tests continue to be plagued by surface pimpling, the causes of which have not yet been determined. Tests seem to indicate that the pimpling is associated with the metal itself, impurities, and grain structure, and is not attributable to poor vacuum or unclean surfaces. Deep-seated damage due to surface grinding may also be a factor. Molybdenum coating for slugs is being investigated. Since no commercial process is available for applying molybdenum, a number of possible methods are being considered, including diffused metal-spray coatings. cementation, electroplating, and vapor plating. A sprayed molybdenum coating has been diffused at 1100°C, producing an alloy coating. Thin and somewhat irregular coatings have been obtained by cementation in molybdenum powder mixed with calcium and magnesium at temperatures of 1000 to 1100°C in combination with a hydrogen treatment at 600°C. Both of these treatments involve temperatures which would cause dimensional changes in slugs and, thus, are of doubtful practical value. No adherent vapor coatings on the base metal have been obtained on account of chloride deposits under the molybdenum. Vapor coating of nickel-plated metal appears to offer promise. (auth)

13790 CT-2483
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS REPORT ON METALLURGY OF TUBALLOY [URANIUM] TO UNIVERSITY OF CHICAGO. Dec. 1, 1944. Decl. Apr. 1, 1957. 24p. $6.30(ph OTS); $3.00(mf OTS).

Dilatometric studies of 0 to 5% Mo alloys indicate that the stable beta region is very small and probably disappears in the neighborhood of 2% Mo. The lower boundary of the stable gamma field drops rapidly with increasing Mo to the neighborhood of 650°C at about 2.5% Mo. Both x-ray and dilatometric data demonstrate the increasing sluggishness of the γ-α transformation with Mo contents above about 2%. Hydrogen solubilities are being studied in 5% Mo alloy in which it has been found that hydrogen diffuses more slowly than in pure U. Dry hydrogen attacks the 5% Mo alloy slowly to form hydride at local areas. The face-centered-cubic phase with $a_0 = 4.93$ A has been proved to be "extraneous," although it has not yet been identified. Pimple formation on metal heated in evacuated glass tubes has been definitely shown to be caused by hydrogen in the original metal. Thorough outgassing of the metal will prevent pimpling. Alloys with 5 and 10% of Co, Ge, Au, Mn, Pd, Ag, and Ti have been prepared and examined microscopically. None of these alloys were single-phase as-cast, the Ag alloys separating into two layers. Heat treatment at 600°C gave a change in structure in the Ti alloys similar to that found for Mo additions, although still showing a second phase. Possible equivalent amounts of Ti on an atomic basis will give similar properties. These alloys were the only ones showing stainless properties. (auth)

13791 FMPC-516(Del.)
National Lead Co. of Ohio, Cincinnati.
PRODUCTION OF ENRICHED SLUGS FOR NORTH AMERICAN AVIATION, INC. Completion Report. H. Carroll, J. Mead, S. L. Reese, and J. Schiltz. Jan. 28, 1955. Decl. with deletions Apr. 24, 1957. 30p. Contract AT(30-1)-1156. $6.30(ph OTS); $3.00(mf OTS).

The production and testing of enriched U slugs are described. Slug specifications are included. (C.W.H.)

13792 HW-35550(Del.)
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
GOLD ALLOYS AS PROCESS-SOLUTION-LUBRICATED SLEEVE BEARINGS. P. B. McCarthy. Mar. 1, 1955. Decl. with deletions Feb. 25, 1957. 17p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Ten materials in which Au was alloyed with Cu, Ag, graphite, Pt, Ni or Zn were evaluated in this study. Corrosion tests to determine the resistance of the alloys to boiling 60 to 65% HNO_3 showed that with the exception of an alloy containing 51 w/o Au and 49 w/o graphite which dissolved in 48 hr and an alloy containing 90 w/o Au and 10 w/o graphite which lost 2.3 w/o in 240 hr the alloys were essentially corrosion resistant. (auth)

13793 HW-39880
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
CONTROLLED OXIDATION OF PLUTONIUM METAL TURNINGS. Robert C. Smith and D. Ross Doman. Nov. 10, 1955. Decl. Mar. 29, 1957. 2p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

A procedure was developed for converting α phase Pu turnings to an oxide suitable for fluorination and reduction to metal. (F.S.)

13794 ISC-507(Del.)
Ames Lab., Ames Iowa.
QUARTERLY SUMMARY RESEARCH REPORT [FOR] APRIL, MAY, AND JUNE 1954. P. Chiotti and O. N. Carlson, comps. Sept. 7, 1954. Decl. with deletions Feb. 26, 1957. 31p. Contract W-7405-eng-82. $6.30(ph OTS); $3.00(mf OTS).

Progress is reported in the following studies: dip coating of Al with Pb; the hot-press bonding of Al to other metals and alloys; thermal cycling of U samples dip coated with Cu–Mg to which Al was hot-press bonded; bonding of Al to Mg; bonding Al to Ni-diffusion-coated U; corrosion resistance of Mg alloys; casting of U into two-mil Zr sleeves; corrosion resistance of U–Zr–Nb alloys; aging of quenched U alloys; phase studies of U–Nb systems; U–Zn liquidus composition; the preparation of Zr–Zn alloys; the etching of U fuel slugs by ionic bombardment; and the preparation of Nb–Zr alloys. (C.H.)

13795 KAPL-1019(Del.)
Knolls Atomic Power Lab., Schenectady, N. Y.
THE EFFECT OF CYCLIC STRAIN ON THE FATIGUE RESISTANCE OF ALPHA-ROLLED URANIUM. L. F. Coffin, Jr. Nov. 30, 1953. Decl. with deletions Feb. 25, 1957. 18p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

The results of an experimental study of the fatigue resistance of alpha-rolled uranium when subjected to cyclic strain at constant temperatures are reported. Strain amplitudes used were sufficiently high to produce failures in the range of 30 to 10,000 cycles. The specimens were aligned both axially and transversely to the axis of the bar from which they were machined and were subjected to uniaxial loading. A significant difference between the longitudinal and transverse fatigue resistance was found. This was attributed to longitudinal stringers of carbides or oxides developed during rolling. The effect was such as to cause a wide scatter in cycles to failure of specimens strain-cycled transversely, such that the probability for fatigue failure in specimens subjected to the strain amplitude and number of cycles of Hanford fuel slugs was finite. Factors such as dimensional instability and irradiation damage are discussed. Fairly good agreement between the experimental results of Stetson and Winton and the results of the present study is found. (auth)

13796 KAPL-M-CME-2
Knolls Atomic Power Lab., Schenectady, N. Y.
COMPARISON OF REQUIREMENTS IN VARIOUS GOVERNMENT PROCUREMENT SPECIFICATIONS FOR STAINLESS STEEL. C. M. Erb. May 23, 1956. 7p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The resumé of differences does not show that substitutions cannot be made, but blanket substitutions cannot be made and each request for substitution requires a knowledge of the specifications. (F.S.)

13797 KAPL-M-CTW-1
Knolls Atomic Power Lab., Schenectady, N. Y.
EVALUATION OF TITANIUM POWDER FOR METALLURGICAL USE. Craig T. Waldo. Apr. 25, 1956. 17p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Six commercially available Ti powders were tested by making small green compacts of the pure metal by cold pressing. These compacts were sintered, hot rolled, pickled, and machined. Bend and tensile tests were then completed to determine the desirability of using these powders in powder metallurgy. (F.S.)

13798 KAPL-M-JMG-3
Knolls Atomic Power Lab., Schenectady, N. Y.
BRAZING TYPE 304 TO TYPE 304 STAINLESS STEEL WITH HIGH TEMPERATURE BRAZING ALLOYS. Final Report. J. M. Gerken. Sept. 13, 1956. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The mechanical behavior of four brazing alloys on type 304 stainless steel in the as-brazed condition, after stress-relief annealing and after exposure to 680°F water corrosion, was investigated. An 82% Au–18% Ni alloy yielded better mechanical properties than the other alloys tested. The ANP-81 alloy (66% Ni–19% Cr–10% Si) was found to be cracked in all samples examined metallographically. None of these alloys were corroded to an appreciable amount in 680°F water. Brittle alloy phases were observed in the joints brazed with ANP-81 and Coast Metals NP (49% Ni–12% Si–4% Mo–28% Fe–5% P). An improvement in the mechanical properties of the ANP-81 braze was apparent after a post heat treatment of 1950°F for two hours. This is believed to be the result of the reduction of brittle phases by diffusion. The Coast Metals NP and Cu brazes were essentially unaffected by this heat treatment. (L.T.W.)

13799 KAPL-M-LFC-1
Knolls Atomic Power Lab., Schenectady, N. Y.
THE STRUCTURAL STABILITY OF FUEL PINS. L. F. Coffin, Jr. May 28, 1951. Decl. Feb. 25, 1957. 14p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

An analysis of the deformation of fuel pins due to sudden power surges is given. The more conventional pressure-diameter change test is converted to a true stress-strain relationship. Instability due to a localized burst of the tube occurs when the wall stress exceeds a value equal to one-half the slope of the stress-strain curve. Effects of cold work and irradiation on the stress-strain curve are discussed. A calculation is carried out for a sudden power increase, in which the pressure and deformation history are computed for various times after the initiation of the surge. (auth)

13800 KAPL-M-RLM-5
Knolls Atomic Power Lab., Schenectady, N. Y.
END CLOSURE DEVELOPMENT FOR A STAINLESS STEEL–25 WEIGHT % URANIUM OXIDE DISPERSION FUEL ELEMENT. R. L. Mehan. Sept. 22, 1955. Decl. Mar. 7, 1957. 25p. Contract W-31-109-Eng-52. $4.80 (ph OTS); $2.70(mf OTS).

Work performed on the development of an end closure at final size for a stainless steel–uranium oxide dispersion element made by the hot double–extrusion method is described. Welded and brazed types of closures were made, tested, and evaluated. On the basis of these tests it was recommended that a nickel manganese brazed square-butt joint be used in any future manufacture of this element. (auth)

13801 KAPL-M-RLM-8
Knolls Atomic Power Lab., Schenectady, N. Y.
THE EFFECT OF HYDROGEN EMBRITTLEMENT ON THE MECHANICAL PROPERTIES OF STEEL; A LITERATURE SURVEY AND REVIEW OF THE PROBLEM. R. L. Mehan. Mar. 27, 1956. 16p. Contract W-31-109-Eng-52. $3.30 (ph OTS); $2.40(mf OTS).

13802 KAPL-M-RNH-1
Knolls Atomic Power Lab., Schenectady, N. Y.
EVALUATION OF LEACHED SPONGE ZIRCONIUM POWDER FROM CARBORUNDUM COMPANY. R. N. Honeyman and J. R. Peloke. May 22, 1956. 4p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

13803 LA-1244
Los Alamos Scientific Lab., N. Mex.
FORMING OF URANIUM IN THE GAMMA PHASE TEMPERATURE RANGE. L. S. Levinson, W. C. Gribble, J. M. Taub, and D. T. Doll. May 1, 1951. Decl. Mar. 26, 1957. 40p. Contract W-7405-eng-36. $6.30(ph OTS); $3.00(mf OTS).

It has been found relatively easy to form U in the γ phase temperature range by hot pressing, forging, or extrusion. The metal is quite plastic and flows readily to form a shape. Several temperatures from 800 to 1000°C were investigated. No forming difficulties were experienced with the metal at the several temperatures concerned. The major difficulty in γ phase hot pressing or extruding was associated with the tools. Metals or ceramics were not successful as tools for one or more reasons concerned with lack of hot strength, reaction with the uranium, failure in thermal shock, and tendency to spall. Graphite was found to be the best material available, but it is not entirely satisfactory because of low strength. Uranium formed in the γ phase possesses some refinement of grain structure as compared with as-cast metal; however, the grain size is quite large. No physical properties of the γ phase formed metal were determined. (auth)

13804 MIT-1045(Pt.II)
Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.
TECHNICAL PROGRESS REPORT FOR THE PERIOD JANUARY THROUGH MARCH 1950. A. R. Kaufmann. May 16, 1950. Decl. Apr. 9, 1957. 13p. $3.30(ph OTS); $2.40(mf OTS).

The oxidation loss of U during rolling has been briefly studied at temperatures between 400 and 600°C. Losses as low as 0.04% were observed at 400°C after a reduction of 50%. Conclusive evidence has been obtained to show that the solubility of U in Pb is small in both the liquid and solid states. The x-ray diffraction patterns used by the British to show large solid solubility is now shown to be due to a compound Pb_3U. (auth)

13805 MIT-1078(Del.)
Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.
TECHNICAL PROGRESS REPORT FOR THE PERIOD

APRIL 1951 THROUGH SEPTEMBER 1951. Apr. 1, 1952. Decl. with deletions Feb. 26, 1957. 114p. Contract AT(30-1)-981. $18.30(ph OTS); $6.00(mf OTS).

Clifton flake Be is considerably finer than Brush QMV in all elements except Cu, Cl, and C. The relatively poor corrosion resistance of Clifton Be in high-temperature water is hard to explain since the chloride content does not correlate with corrosion. Open pot melting of Be was again explored in order to study the effect of pouring temperature on ingot grain size. No remarkable results were obtained. Forty-pound castings of Zr–Sn alloys have been made using graphite crucibles and self-pouring. Carbon contents as low as 0.08% were obtained and the Sn is uniformly distributed. Zr and Ti pressure welds made in vacuum can be as strong as the annealed metal. Temperatures of 1200 and 1000°C, respectively, are required with application of about 1000 psi during welding. The grain size of Zr air-cooled from the beta phase must be described in terms of both the beta grain size and the Widmannstatten structure of alpha metal. It appears that oxygen refines both of these grain sizes, while hydrogen refines only the alpha structure. Addition of oxygen to Zr up to 0.3% raises the yield point markedly at room temperature but not at 300°C. Hydrogen in amounts up to 20 at. % causes considerable strengthening at room temperature with appreciable ductility remaining. Most of this hydrogen is present as a second phase. A tentative U–Zr phase diagram is presented. It is believed that the body-centered-cubic phase extends from gamma U to alpha Zr without interruption and that an intermetallic phase, of variable composition and decomposing at about 600°C, exists in the middle of the system. The present status of the Zr analytical program is described. The use of powdered ZrH_2 for a standard sample has met with general approval. The technique for making the hydride is presented. A careful study of the slip casting of beryllia to yield high density bodies has been made. The technique is described in detail. Apparatus for hot pressing beryllia-beryllium is in operation. A technique for brazing of graphite with Zr is described. Laminates of graphite brazed in this way show remarkable strength, particularly if the surface has a Zr coating. (auth)

13806 NMI-1121

Nuclear Metals, Inc., Cambridge, Mass.

DOW-DETROIT EDISON PROJECT PROGRESS REPORT, JULY 1 TO OCTOBER 1, 1954. R. O. Evans and A. Boltax. Nov. 12, 1954. Decl. Mar. 22, 1957. 32p. Contract AT(30-1)-1565. $6.30(ph OTS); $3.00(mf OTS).

Investigation conducted to develop and perfect the methods and techniques necessary to produce a workable core element for the Dow-Detroit Edison Power Project are reviewed. The methods under consideration include U alloy castings about a radiator core, cast flat plates, and hot-worked, corrugated plates. U-Cr alloy is being cast around Fe and Zr radiator assemblies with 0.165 in. O.D., 0.145 in. I.D. tubing, spaced 0.195 in. on centers. This same alloy is being used for flat plate castings, which are aimed at making 0.040 in. thick plate, 3.5 in. wide and 27 in. long. Because of its eutectic nature and low melting point, the U-Cr alloy has been used in preliminary hot forging work also. All elements now contemplated with U-Cr alloy would probably employ a stagnant sodium film to effect heat transfer between fuel and coolant tubing. The U-Zr alloy has been cast around Zr tube radiators in an attempt to obtain a metallurgical bond between fuel and tube for the purpose of high heat transfer properties. (auth)

13807 SEP-179

Sylvania Electric Products Inc. Atomic Energy Div., Bayside, N. Y.

ZONE MELTING OF URANIUM. C. I. Whitman, V. Compton, and R. B. Holden. June 15, 1955. Decl. Mar. 7, 1957. 20p. Contract AT-30-1-GEN-366. $0.20(OTS).

A study was made of the possible application of the zone melting technique to the separation of U from fission products and other impurities. Bar specimens of U having various alloying elements were zone purified in thoria-coated alumina boats. Uranium bars previously irradiated in the Brookhaven pile were also zone melted. Although zone melting did not appear promising as a method of processing irradiated U because of slagging of fission products into the surface of the bar, this technique has potential application to removing such fission products as Zr, Nb, and Ru, and to purifying ordinary U from such impurities as B, Fe, Si, Ni, and Co. (auth)

13808 TID-5243(Rev.)

[Metallurgy Development Advisory Committee, AEC.]

TECHNICAL PAPERS OF THE EIGHTH METALLOGRAPHIC GROUP MEETING HELD AT THE OAK RIDGE NATIONAL LABORATORY, DECEMBER 9 AND 10, 1953. H. P. Roth, comp. Decl. Mar. 1, 1957. 64p. $0.50(OTS).

The papers presented at the Eighth Metallographic Group Meeting are compiled. Abstracts are given rather than complete texts when the paper has been included in some previous publication. Part I consists of 8 papers dealing with inclusions in U. In some cases, metallographic examination has included chemical or mechanical removal of inclusion particles with subsequent examination of these particles by x-ray diffraction, chemical, or spectrographic analysis. Among the inclusions encountered were MgF_2, UO, UO_2, UN, UC, UC_2, UH_3, and solid solution particles of UC–UO–UN. Part II includes a report on the preliminary results of cooperative effort on the identification of inclusions in U, a report on auxiliary technique for remote metallography, and supplemental information on specimen mounting in casting resin. (J.E.D.)

13809 WAPD-FE-62

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

A PROCEDURE FOR THE FABRICATION [OF] CLAD PELLETS OR RODS BY EXTRUSION. J. Halapatz. Apr. 21, 1954. Decl. Mar. 19, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

Billet designs are proposed which, in place of cylindrical fuel loading, utilize radial or offset loading, thereby providing in the extrusion billet core discontinuities necessary for end cladding. (L.M.T.)

13810 WAPD-FE-68

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

AN EVALUATION OF EXTRUSION BILLET DESIGN–STEEL JACKETING VS. BARE EXTRUSIONS. Report on Experiment PWR 11d1. J. Halapatz. Apr. 26, 1954. Decl. Apr. 15, 1957. 9p. $1.80(ph OTS); $1.80(mf OTS).

Fuel bearing rods with Zircaloy-2 clad but without Ti–Namel jacketing were successfully extruded. Welding box assembly of extrusion billets in a helium atmosphere is considered a feasible technique. (auth)

13811 WAPD-FE-71

[Westinghouse Electric Corp.] Atomic Power Div., Pittsburgh.

THE FABRICATION OF CLAD URANIUM–MOLYBDENUM FUEL BEARING RODS FOR METALLURGY EXPERIMENTS. Report of Experiment PWR 11E1. J. Halapatz. Apr. 29, 1954. Decl. Apr. 15, 1957. 13p. $3.30(ph OTS); $2.40(mf OTS).

The experimental fabrication of Zircaloy-2 clad rods containing U–Mo alloy cores is described. Extrusion to final fuel element dimensions is not recommended.

Drawing of extruded composite rods is proposed to obtain more accurate dimensioning. (auth)

13812 WAPD-FE-77

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

THE PRESSURE BONDING OF NATURAL URANIUM IN ZIRCALOY-2 HEMISPHERICAL CUPS. B. E. Schaner. May 8, 1954. Decl. Apr. 15, 1957. 9p. $1.80(ph OTS); $1.80(mf OTS).

Experiments show that U-filled pellets can be produced by pressure bonding U cores to prefabricated Zircaloy-2 shells. These pellets could be made with either solid or hollow cores. Cores need not be preformed, but may be formed during the bonding operation. The time required for this forming and bonding operation could be reduced considerably by using automatic feed of components and by utilizing induction techniques for heating. (auth)

13813 WAPD-FE-230

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

ZIRCALOY-2 CLAD BI-METAL FUEL EXTRUSION. Report on Experiment 11aj(1). R. E. Droegkamp. Nov. 4, 1954. Decl. Mar. 20, 1957. 6p. $1.80(ph OTS); $1.80 (mf OTS).

An attempt to extrude rods containing 3.8 wt. % Si–U fuel separated from a Zircaloy-2 cladding by a thin layer of a 12 wt. % Mo–U alloy resulted in the eruption of the fuel through the Zircaloy-2 cladding almost continuously along the length of the rod. (auth)

13814 WAPD-FE-231

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

INVESTIGATION OF BARRIER LAYERS FOR EXTRUDED ZIRCALOY-2 CLAD MOLYBDENUM–URANIUM RODS. Report on Experiment 11f(1). R. E. Droegkamp. Nov. 4, 1954. Decl. Mar. 20, 1957. 12p. $3.30(ph OTS); $2.40 (mf OTS).

Zircaloy-2 clad 12 wt. % Mo–U rods having Mo, Ta, sponge Zr, and Nb barrier layers between the clad and fuel with final thicknesses of from 0.3 to 2 mils were extruded from a 1.6-in. container to 13:1 and 25:1 reductions. Except for the Mo, all barrier layers appeared to be reasonably uniform in thickness. All showed intimate contact with the fuel and clad with no apparent diffusion. (L.M.T.)

13815 WAPD-FE-1036

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]

K FACTORS OF VARIOUS MATERIALS BASED ON STARTING LOADS. R. W. Tombaugh. Sept. 21, 1955. Decl. Apr. 15, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

The resistance to deformation (by extrusion) is given for Cu, Hf, Pb, Mo, Mo–Co alloy, steels, U alloys, Zircaloy-B and -2, and Zr. (M.H.R.)

13816 WAPD-FE-1071

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

THE FABRICATION OF ENRICHED FUEL RODS FOR THE WAPD-NRX-CR-5 TEST. J. G. Goodwin, W. B. Haynes, and E. L. Richards. Oct. 13, 1955. Decl. May 1, 1957. 12p. Contract AT-11-1-GEN-14. $3.30(ph OTS); $2.40 (mf OTS).

The fabrication of enriched-U–12 wt. % Mo fuel rods for testing under pressurized water reactor conditions is described. (T.R.H.)

13817 WAPD-LSR(MM)-43

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

THE OXIDATION OF A ZIRCONIUM – 50 w/o URANIUM ALLOY IN OXYGEN. L. D. Kirkbridge and D. E. Thomas. Sept. 26, 1955. Decl. Mar. 21, 1957. 11p. $0.25(OTS).

The oxidation studies were made on U–Zr specimens machined from strip stock which was previously hot rolled at 1450°F. Oxidized specimens were sectioned and examined by x-ray techniques. Corrosion data are included, and the phases present as a function of distance from the oxide–gas interface are tabulated. Equations are given for the linear corrosion characteristics. The investigation showed that the oxidation product is a tetragonal structure below 400°C and that the effect of 0 on the ϵ phase is pronounced from 750 to 900°C. (B.J.H.)

13818 WAPD-MM-287

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]

RÉSUMÉ OF URANIUM ALLOY DATA. B. Lustman. Feb. 19, 1954. Decl. Mar. 22, 1957. 25p. $0.35(OTS).

Additional data on uranium alloys which have been compiled are summarized. Of greatest significance probably are the results of preliminary irradiation tests on the U–Mo alloys and the results of cathodic treatment of clad and unclad samples of U–Mo alloys. Irradiation of an 11.5% Mo–U alloy to approximately 3000 Mdw/t burn-up revealed no change in the general corrosion resistance of the alloy on subsequent corrosion testing but a marked decrease in the time for discontinuous failure. Cathodic treatment of unclad U–Mo alloys in boiling water revealed formation of uranium hydride needles in the alloy matrix. The time for discontinuous failure of samples thus treated on subsequent high-temperature corrosion testing is drastically reduced over that of untreated samples. Cathodic treatment of "defected" clad samples, however, revealed no pick-up of hydrogen in the form of hydride needles. (auth)

13819 WAPD-MP-244

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PRESSURE WELDING OF METALS. H. R. Hoge. Mar. 11, 1953. Decl. Mar. 19, 1957. 12p. Contract AT-11-1-GEN-14. $3.30(ph OTS); $2.40(mf OTS).

A literature survey regarding bonding in several common metals is presented. (T.R.H.)

13820 WAPD-PMM-15

Westinghouse Electric Corp. Atomic Energy Div., Pittsburgh.

RESUME' OF URANIUM ALLOY DATA–VIII. B. Lustman, Feb. 25, 1955. Decl. Mar. 20, 1957. 83p. $13.80(ph OTS); $4.80(mf OTS).

The effects of melting, extrusion temperature, homogenization, and other variables on the corrosion properties of U–Mo were studied. The corrosion resistance of U–Mo–Pt, U–Mo–Nb, U–Mo–X (where X is Cr, Ti, or V), and U–Zr–Nb to high-temperature water and the irradiation behavior of U–Mo and U–Si were investigated. Work continued on the preparation, powder metallurgy, and irradiation testing of UO_2. (C.W.H.)

13821 WAPD-RM-215

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

IMPACT TESTS OF ZIRCALOY 2-HAFNIUM WELDS. Harry R. Hoge. Dec. 30, 1953. Decl. Mar. 9, 1957. 12p. $3.30(ph OTS); $2.40(mf OTS).

The impact data obtained for alpha annealed Zircaloy 2–Hf welds indicate that at room temperature the welded material is similar to unwelded alpha annealed Zircaloy 2. At operating temperatures the welds are as ductile as Zircaloy 2 (none of the samples broke at 500°F) despite the number of gas pockets in the weld metal. It was also shown that all of the materials are stronger than crystal bar Zr. No particular advantage would be gained by heat

treating the Allegheny Ludlum Zircaloy 2 control rod adapter shock prior to welding to Hf. (auth)

13822 WAPD-T-46

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

MECHANICAL PROPERTIES AND MICROSTRUCTURE OF ANNEALED ZIRCONIUM. W. L. Mudge, Jr. June 10, 1953. Decl. Mar. 7, 1957. 25p. Contract AT-11-1-GEN-14. $4.80(ph OTS); $2.70(mf OTS).

The mechanical properties of unalloyed Zr are shown to vary markedly depending on the state of anneal of the metal. Hardness is used as an index of annealing condition. Precautions to be taken in obtaining reproducible hardness values in Zr are outlined. In addition, the appendix contains a hardness conversion table for Zr base alloys. This covers the hardness range from approximately 60 to 250 DPH (20 to 100 R_B). (auth)

13823 WAPD-T-185

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

THE APPLICATION OF CATHODIC VACUUM ETCHING TO ELECTRON METALLOGRAPHY. T. R. Padden. [1954?]. 18p. Contract AT-11-1-GEN-14. $3.30(ph OTS); $2.40(mf OTS).

A cathodic vacuum etching technique which has been used extensively for light microscopy is shown to be suitable for electron microscopy. It develops the fine structure of the materials tested as well as the general microstructure. In general, it produces etched surfaces which show no indication of redeposited cathode material and are apparently free from stains at the level of resolution of electron metallography. (auth)

13824 WASH-125

Division of Research, AEC.

URANIUM ALLOY NEWSLETTER NO. 4. David W. Lillie, ed. Aug. 1953. Decl. Mar. 7, 1957. 43p. $6.30(ph OTS); $3.00(mf OTS).

Armour Research Foundation. Development of a U-base alloy that is corrosion resistant in 600°F water is reported. Battelle Memorial Inst. Investigations of phase diagrams, heat treatment of U–Zr alloys, transformation kinetics of low-melting U alloys, effect of Fe additions upon the structure and allotropic transformation temperatures of the Cr–U eutectic alloy, effect of C on the dimensional stability of U, corrosion of U–Zr alloys, and the creep properties of U are presented. Knolls Atomic Power Lab. Radiation effects on C–U alloy, heat treatment and physical properties of Cr–U alloys, and corrosion resistance of U–Zr alloys to 800°C water are reported. Massachusetts Inst. of Tech. Preliminary experiments were made to determine the resistance of W, Ta, and Zr to liquid eutectic C–U. National Bureau of Standards. The essential features of the Ag–U system, thermal analysis of the Au–U system, and preliminary studies on the transformation kinetics of the Ti–U system are outlined. New York Operations Office. Development of U–Zr alloys by co-reduction and production of Mo–U alloys is reported. Sylvania Electric Co. Cold pressing and sintering of Nb–U alloys, and the diffusion and thermal cycling of Al–U, Mo–U, Si–U, and Bi–U alloys are reported. Westinghouse Atomic Power Div. A brief study is presented of the effects of alloying additions on the corrosion resistance of ternary U alloys. (J.E.D.)

PHYSICS

13825 CF-56-12-118

Metal Hydrides Inc., Chemical Research Lab., Beverly, Mass.

PERMEATION OF HYDROGEN THROUGH MATERIALS AND HYDROGENOUS SYSTEMS FOR SHIELDING. Quarterly Status Report No. 1 [for] July 1, 1956 through September 30, 1956. Edward A. Sullivan and Sidney Johnson. Dec. 10, 1956. Decl. Mar. 9, 1957. 23p. [For Oak Ridge National Lab. Contract W-7405-eng-26, Subcontract 390.] $4.80(ph OTS); $2.70(mf OTS).

Rates of H_2 permeation through molten NaOH, NaOH + Ni, and Cu are given. The aminolysis of Li borohydride is discussed, with effects of temperature, composition, concentration, etc., considered. (L.M.T.)

13826 CRD-T2A-41

California Research and Development Co., Livermore, Calif.

NEUTRON PRODUCTION COMPARISON [OF] WATER-COOLED ZIRCONIUM CLAD URANIUM; NaK-COOLED URANIUM; AND NaK-COOLED THORIUM. R. L. McKisson. Aug. 17, 1951. Decl. Feb. 15, 1957. 11p. (LWS-23073). $3.30(ph OTS); $2.40(mf OTS).

A 200% increase in neutron production would be expected with a substitution of U for Th in the NaK cooled plate target. A 12% increase in neutron production would be expected from the use of a water cooled Zr clad U target instead of a NaK cooled Th target. However, U plates can be considered only if dimensional stability can be obtained. (F.S.)

13827 CRD-T4A-75

California Research and Development Co., Livermore, Calif.

THERMAL AND RESONANCE UTILIZATION CALCULATION METHODS FOR SLAB GEOMETRY. D. H. Imhoff. [Nov. 11, 1951]. Decl. Apr. 2, 1957. 4p. (LWS-24171). $1.80(ph OTS); $1.80(mf OTS).

13828 CRD-T4A-126

California Research and Development Co., Livermore, Calif.

EXTRAPOLATION OF NEUTRON YIELDS WITH INCREASING PARTICLE ENERGY. D. H. Imhoff and C. C. Old. July 11, 1952. Decl. Apr. 2, 1957. 8p. (LWS-24442). $1.80(ph OTS); $1.80(mf OTS).

An extrapolation with energy is made by two different approaches in the hope that the limits of neutron yields from deuteron and proton bombardment of solid U targets might be established. (T.R.H.)

13829 CRD-T4A-140

California Research and Development Co., Livermore, Calif.

THE TRANSPORT OF NEUTRONS IN CYLINDRICAL CAVITIES. W. H. Harker. July 23, 1952. Decl. Apr. 2, 1957. 14p. (LWS-24478). $3.30(ph OTS); $2.40(mf OTS).

Methods are developed which will yield information concerning variation of neutron flux and escape of neutrons from the target assembly of the MTA. (T.R.H.)

13830 IDO-16360

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

SAFETY OF MTR FUEL SHIPPING BOXES. H. L. McMurry and B. L. Hanson. Oct. 15, 1956. 15p. Contract AT(10-1)-205. (PTR-111). $3.30(ph OTS); $2.40(mf OTS).

MTR fuel is now surrounded by rubber padding and shipped in wooden boxes. The possibility of getting a critical assembly of these boxes is examined. Conclusions and recommendations are: (1) As few as 12 boxes might form a critical array. (1) Subcriticality can probably be assured by putting a Cd plate in the top of each fuel box. (3) The only certain safety analysis in this and similar cases is to measure the multiplication as a function of the number of boxes assembled in a group. (auth)

13831 K-1260
Oak Ridge Gaseous Diffusion Plant, Tenn.
CRITICALITY CALCULATIONS FOR HYDROGENOUS SYSTEMS. J. R. Knight. Nov. 25, 1955. Decl. Mar. 14, 1957. 20p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

A theoretical method of calculating the critical mass of unreflected hydrogenous systems containing U^{235}, based on a spherical harmonics expansion for hydrogen moderation and on age theory slowing down for all other materials, is developed and set up for machine computation on the IBM 607. The calculated critical masses for high assay systems agree with experimental values within 13% for hydrogen to U^{235} ratios from 40 to 750, and agree within 12% for a single calculation at 4.9% U^{235} assay and a hydrogen to U^{235} ratio of 145. The calculating method appears to yield results which are directly applicable in the evaluation of the nuclear safety of uranium processing equipment. (auth)

13832 K-1317
Oak Ridge Gaseous Diffusion Plant, Tenn.
SELF-CONSISTENT CRITERIA FOR EVALUATION OF NEUTRON INTERACTION. H. F. Henry, C. E. Newlon, and J. R. Knight. Dec. 21, 1956. Decl. Mar. 15, 1957. 24p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

New safe interaction criteria for containers of fissionable materials handled at the Oak Ridge Gaseous Diffusion Plant have been developed on the basis of an interaction theory using the basic concepts of a safe solid angle subtended by interacting containers, and the multiplication factor as determined by two-group theory for an individually safe container. The calculated results agree satisfactorily with experimental data obtained with identical interacting units involving both cylinders and slabs containing highly enriched uranium, the core compositions of which were varied between H—U^{235} atomic ratios of 44.3 and 337. The application of the derived interaction criteria to items containing material with low moderation or low U^{235} assay, and to containers for which nuclear safety is dependent upon control of the U^{235} mass or U^{235} concentration is discussed. (auth)

13833 KAPL-335
Knolls Atomic Power Lab., Schenectady, N. Y.
A SUGGESTION CONCERNING THE DIRECT GENERATION OF ELECTRICITY FROM FISSION ENERGY. K. H. Kingdon. May 1, 1950. Decl. Mar. 27, 1957. 6p. Contract W-31-109-Eng-52. $0.20(OTS).

A device for the direct generation of electricity from fission energy is proposed. A mixture of enriched UF_6 and F_2 in which a chain reaction is proceeding is considered. Fissions in the gas will raise the electrons, the ions, and the gas to a very high temperature. At the center of the device is a cylindrical core of fissionable materials which is slightly subcritical. This core is surrounded by a cylindrical jacket containing a coolant. Surrounding the central cylinder is a large spherical gas blanket made up of a mixture of UF_6 and F_2 contained in a monel sphere. Operation of the device is described. (L.T.W.)

13834 KAPL-1515
Knolls Atomic Power Lab., Schenectady, N. Y.
REPORT OF THE PHYSICS SECTION FOR DECEMBER, 1955, JANUARY, FEBRUARY 1956. Decl. Apr. 9, 1957. 33p. Contract W-31-109-eng-52. $0.40(OTS).

The preliminary performance of a three-stage mass spectrometer is discussed. A block diagram of the spectrometer is shown, and a spectrum of enriched U obtained with this machine is given. Micrographs are given of a LiF single crystal disk irradiated at 30°C to 2.4×10^{18} neutrons/cm^2 and of a hot-pressed polycrystalline cylinder before and after irradiation. X-ray-diffraction studies of radiation damage are also discussed. The nuclear constants of Pu are discussed. Other experiments in nuclear physics are summarized. (For preceding period see KAPL-1468.) (B.J.H.)

13835 KAPL-M-HH-6
Knolls Atomic Power Lab., Schenectady, N. Y.
SAFETY CALCULATIONS FOR UNLOADING TUNNEL AND STORAGE VAULT. H. Hurwitz, Jr. Feb. 9, 1949. Decl. Mar. 13, 1957. 5p. Contract [W-31-109-Eng-52]. $1.80 (ph OTS); $1.80(mf OTS).

The design of the unloading tunnel and fuel storage vault was studied to make sure that no possibility of a chain reaction exists. It is concluded that it is safe to store 100 kg of U^{235} in fuel rods in a rack 6 ft × 20 ft if the vault cannot be flooded, and it is probably safe even if it were to be flooded. (M.H.R.)

13836 ORNL-1345(Del.)
Oak Ridge National Lab., Tenn.
ELECTROMAGNETIC RESEARCH DIVISION QUARTERLY PROGRESS REPORT, PART I, FOR PERIOD ENDING JUNE 30, 1952. F. T. Howard, ed. Nov. 14, 1952. Decl. with deletions Mar. 1, 1957. 23p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

On the 86-inch cyclotron a 41-kw beam was calorimetered at a net ion loading efficiency of 40%; the average proton current was 1.85 ma at 22.5 Mev. Practical specific yields have been determined for (p,2n) reactions on Zn and Bi. The investigation of products of proton-induced fission of U has been continued and new techniques are being used in measuring angular distribution of reaction products. The 63-inch heavy particle cyclotron is now in operation; N^{3+} particles have been accelerated to ~25 Mev. In preliminary tests, induced activities have been detected in C, N_2, and O_2 targets. A hot-cathode type ion source is ready for test operation. The 22-inch cyclotron is being used in an investigation of the problems associated with the use of rf and dc electrodes for the acceleration of protons from the ion source into the dees. Radiation-induced corrosion in Inconel tubing containing #21 eutectic was produced by proton irradiation in the 86-inch cyclotron; it is shown that the corrosion was not due to thermal effects. Approximately 150 g of highly purified U^{238} (<5 ppm U^{235} have been prepared, and two grams of Th^{230} was enriched to 90.6%. (auth)

13837 UCRL-1680(Del.)
California. Univ., Berkeley. Radiation Lab.
MTA QUARTERLY PROGRESS REPORT FOR SEPTEMBER, OCTOBER, NOVEMBER 1951. Feb. 29, 1952. Decl. with deletions Mar. 5, 1957. 127p. Contract W-7405-eng-48. $19.80(ph OTS); $6.30(mf OTS).

Progress in the following phases of the MTA program is reported: cavity design; x-ray and sparking tests in a Hg-pumped system; x-ray monitored vacuum RF sparks; DC drain and breakdown phenomena for unoutgassed metals; ion pump development; Mark I injector; injector electrical equipment; high-frequency program; Mark I electronics studies; drift-tube tests; MTA mechanical design; Mark I targets, including metallurgy, thermal cycling, corrosion, and Po escape; A-12 target and lattice development, including target design and development, materials research, lattice calculations, target neutron leakage, internal neutron flux distribution for solid U target, nuclear physics and chemistry, fission and capture in Mark II targets, isotopic composition of product in primary and secondary targets; fission yields as a function of neutron energy, deuteron-produced activities in coolants, diffusion- and corrosion-introduced

activities, and instrumentation; MTA chemical processing; theoretical studies; electron-model Thomas cyclotron; and 20-in. cyclotron project. (D.E.B.)

13838 UCRL-1903(Del.)
California. Univ., Berkeley. Radiation Lab.
MTA QUARTERLY PROGRESS REPORT [FOR] MARCH, APRIL, AND MAY 1952. Aug. 7, 1952. Decl. with deletions Mar. 5, 1957. 183p. Contract W-7405-eng-48. $28.80(ph OTS); $8.40(mf OTS)

Progress is reported on the following phases of the MTA project: cavity design, sparking and x rays in a Hg-pumped vacuum system, drift tube cleaning technique, spark phenomena and r-f decay of the Livermore accelerator, ion pump program, h-f program, injector electrical equipment, MTA mechanical design, MTA target and lattice studies, MTA chemical process studies, nuclear chemistry, cyclotron studies, and the 20-in. cyclotron project. (F.S.)

13839 WAPD-A1W(M)-307
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
MTR A-33 PRESSURE TUBE FEASIBILITY STUDY. M. Doumas and W. R. Gibson. Apr. 4, 1956. 16p. $3.30 (ph OTS); $2.40(mf OTS).

A tube for proof testing Carrier Vessel Reactor and Shippingport Pressurized Water Reactor subassemblies in the MTR is described and its feasibility studied. (T.R.H.)

ELECTRICAL DISCHARGE

13840 LA-1944
Los Alamos Scientific Lab., N. Mex.
STATUS REPORT ON THE K-4 MAGNETIC INDUCT ON MACHINE. L. M. Bagget, T. L. Franklin, and A. Van Duren—Vernal Josephson, ed. Aug. 8, 1955. Decl. Apr. 2, 1957. 28p. Contract W-7405-eng-36. $4.80(ph OTS); $2.70(mf OTS).

Due to laboratory interest in devices capable of production of high current with short rise times in gaseous discharges (plasmas), an experimental machine has been built and tested to obtain data applicable to the design of a high gradient magnetic induction machine. This machine consists of a condenser type (19 μf at 25 kv) energy storage bank air core coupled to a toroid (~ 1 meter path length) in which the plasma is produced. It has been used to determine circuit parameters and the effect of these parameters on the plasma. Results obtained thus far indicate that for a plasma in this system the minimum value of resistance is ~ 0.07 ohm, and this value might be dependent on the parameters of the machine, i.e., source impedance, coupling, leakage inductance, etc. (auth)

INSTRUMENTS

13841 CF-56-2-168
Oak Ridge National Lab., Tenn.
PRESSURE MEASUREMENT BY DISPLACEMENT OF A RESTRAINED DIAPHRAGM. D. M. Richardson. Feb. 15, 1956. 16p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A relatively simple method of determining the pressure of enclosed systems by the displacement of a restrained diaphragm was developed. The method was used to measure unsaturated steam pressure in connection with volumetric water adsorption studies, but is applicable to other systems. (D.E.B.)

13842 KLX-1042
Kellex Corp., New York, New York.
REPORT OF THE INSTRUMENT AND MECHANICAL DEVELOPMENT DEPARTMENT ON REMOTE SAMPLING APPARATUS. Job. 15. June 30, 1949. Decl. Feb. 25, 1957. 34p. Contract W-31-109-eng-52, Subcontract G-148. $0.40(OTS).

This report is one of a group of 11 Instrument and Mechanical Development Department status reports presenting detailed information on the investigations and work summarized in KLX-1033, "Summary Report, Research and Development Program." The program herein summarized describes the efforts of this group in developing a remote device for obtaining representative samples of highly radioactive process solutions. The problem was complicated by the necessity for complete personnel protection by shielding and complete elimination of the possibility of drippage. The In-Line Sampler has been designed around a special plug-type valve with a stainless-steel plug seated in a Teflon liner. This sampler is very promising. (a''h)

ISOTOPE SEPARATION

13843 A-781
Columbia Univ., New York. Div. of War Research.
DESIGN OF CASCADES OF THERMAL DIFFUSION COLUMNS FROM EXPERIMENTAL DATA: A NOMOGRAPH OF RAPID CALCULATIONS. Karl Cohen. Aug. 13, 1943. Decl. Jan. 31, 1957. (4-R-95); Includes Attachments: MEMORANDUM ON THE THERMAL DIFFUSION METHOD AS CARRIED OUT BY THE NAVAL RESEARCH LABORATORY. Harold C. Urey. Feb. 19, 1943. (100M-79); [Letter to W. K. Lewis, Re: NAVAL RESEARCH LABORATORY METHOD.] Harold C. Urey. Sept. 4, 1943. (100U-L-260); [Letter to Lyman J. Briggs, Re: THERMAL DIFFUSION METHOD.] Harold C. Urey. Jan. 28, 1943. (100L-39); [Letter to James B. Conant, Re: LIQUID THERMAL DIFFUSION PROCESS.] Lyman J. Briggs, H[arold] C. Urey, E. V. Murphree, and W. K. Lewis. Sept. 8, 1943; and [Letter to James B. Conant, Re: LIQUID THERMAL DIFFUSION PROCESS AS AN ALTERNATE TO THE DIFFUSION SCREEN PROCESS.] Lyman J. Briggs. Jan. 30, 1943. 15p. Project SAM. Contract W-7405-eng-50. $3.30(ph OTS); $2.40(mf OTS).

Some considerations preliminary to the design of cascades for thermal diffusion separation of U isotopes are given. (T.R.H.)

13844 KAPL-M-LMD-2
Knolls Atomic Power Lab., Schenectady, N. Y.
ISOTOPE SEPARATION BY THE PHOTOEXCITATION OF SPECIFIC ISOTOPES; THE POSSIBILITY OF APPLYING THIS METHOD TO THE SEPARATION OF Li^6 AND Li^7. Leon M. Dorfman. Nov. 28, 1952. Decl. Mar. 22, 1957. 5p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80 (mf OTS).

13845 M-5507
Girdler Corp. Gas Processes Div. Process Development and Research Labs., Louisville, Ky.
SEPARATION OF THE BORON ISOTOPES BY ATMOSPHERIC DISTILLATION OF DIMETHYL ETHER-BORON TRIFLUORIDE COMPLEX. W. M. Keely and Kenton Atwood. June 3, 1953. Decl. Mar. 6, 1957. 25p. Contract [AT(30-1)-1506]. (T9.40-2-2). $7.80(ph OTS); $3.30(mf OTS).

13846 TID-452
Oak Ridge National Lab., Tenn.
ELECTROMAGNETIC RESEARCH DIVISION MONTHLY INFORMATION MEETING, SEPTEMBER 15, 1950. R. S. Livingston, [comp.]. Sept. 15, 1950. Decl. Mar. 1, 1957. 35p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00 (mf OTS).

The large alpha and beta magnets, 86-in. cyclotron, and 22-in. cyclotron at Y-12 lab in Oak Ridge are described. Some experiments with separation of U isotopes are briefly described. (T.R.H.)

MASS SPECTROGRAPHY

13847 K-1289
Oak Ridge Gaseous Diffusion Plant, Tenn.
LINE RECORDER MASS SPECTROMETER OPERATION WITH AN OIL DIFFUSION PUMP AND CARBON TRAP. W. P. McEvoy and B. J. Bogardus. June 25, 1956. Decl. Feb. 25, 1957 17p. Contract W-7405-eng-26. $0.30(OTS).

Tests were conducted on a Line Recorder Mass Spectrometer Tube Rack, with a Litton oil diffusion pump and an activated carbon trap, to ascertain the feasibility of replacing the mercury diffusion pump and mechanically refrigerated cold trap. Mass spectrum scans are included which show the residuals in the spectrometer analyzer with the oil diffusion pump and carbon trap and also with the mercury diffusion pump and mechanically refrigerated cold trap. A comparison of the holding pressures of the Litton Model PB Oil Vapor Pump and the mercury diffusion pump is presented with the effect of the fore vacuum pressure on residual gas ion signals. The back diffusion of the silicone fluid vapor into the spectrometer source was not completely eliminated by the activated carbon trap. The characteristic ion peaks of Dow Corning Silicone Type 703 fluid interfere with the quantitative analysis of the various contaminants encountered in the routine monitoring of the cascade. Such a trap is not recommended as a substitution for the existing cold trap. (auth)

13848 KAPL-M-RES-1
Knolls Atomic Power Lab., Schenectady, N. Y.
ESTIMATED AMOUNT OF SAMPLE NECESSARY FOR A NORMAL MASS SPECTROMETRIC ANALYSIS. R. E. Schofield. Sept. 9, 1949. 5p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

13849 TID-5423(Del.)
Union Carbide Nuclear Co. K-25 Plant, Oak Ridge, Tenn.
LABORATORY ISOTOPE MASS SPECTROMETER. Job Specification JS-747, Rev. 2. Sept. 16, 1955. Decl. with deletions Mar. 21, 1957. 47p. Contract [W-7405-eng-26]. (KD-630-2(Del.)). $9.30(ph OTS); $3.60(mf OTS).

The design of an isotope mass spectrometer is presented, which is used for the ratio measurements of U^{235} and U^{238} in UF_6 samples. (W.L.H.)

MATHEMATICS

13850 KAPL-M-EBF-6
Knolls Atomic Power Lab., Schenectady, N. Y.
USE OF IBM-650 IN REDUCTION OF FOIL COUNTING DATA. E. B. Fehr. Jan. 10, 1955. 19p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

Subroutines were developed to aid in calculations for processing data obtained in foil activation experiments by the IBM-650 digital computer. An automatic counting system was developed to accommodate the large increase in experiments resulting from measurements of power distribution and flux gradients. (F.S.)

13851 ORNL-1290(Del.)
Oak Ridge National Lab., Tenn.
MATHEMATICS PANEL QUARTERLY PROGRESS REPORT FOR PERIOD ENDING APRIL 30, 1952. C. L. Perry, ed. July 24, 1952. Decl. with deletions Mar. 1957. 27p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

Brief statements are presented of the background and status of the following projects on which the Mathematics Panel is working: methods of computation for use with a high-speed automatic-sequenced computer; calculation of percentage points of the Logarithmic normal distribution; test of a probability model for cell formation in neurospora conidia; oxygen and temperature effects on radiation-induced lethal mutations in drosophile melanogaster; derivation of a likelihood ratio test for a null hypothesis in an irradiation experiment with corn pollen; estimation of volume of lymph space; comparison of x and β irradiation effects on chromosome fragmentation; heat effect on cell extinction; kinetics of the $HBr-HBrO_3$ reaction; professional selection program; numerical evaluation of integrals; neutron penetration in a 5-cm slab of paraffin; thermal neutron damage; evaluation of the lateral distribution of energy dissipated by a moving ion; some parameter estimates in a near water solution; estimation of total proton track length in photographic emulsion; beta decay (field factors); neutron decay theory; energy levels of nuclei; calculation of internal conversion coefficients with screening; determination of the fast neutron flux of the X-10 graphite pile; calculation of Racah coefficients for the angular distribution in nuclear reactions; T + D and He^3 + D cross sections; mixed internal conversion correlation coefficients; reactor response curves; machine computation of solutions for the neutron transport equation; miscellaneous computations for ANP Project; depigmentation as a biological dosimeter—operation greenhouse; two-group three-region computations for a production reactor; bulk shielding facility; kinetics of HRE; characteristics of explosion and detonation of combustible gases generated by the ISHR; boiling reactor studies; kinetics of boiling reactors; circulation in boiling reactors. (D.E.B.)

MEASURING INSTRUMENTS AND TECHNIQUES

13852 HW-41412
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
OPERATION AND MAINTENANCE OF THE HANFORD EFFLUENT WATER GAMMA MONITOR. M. C. Greene, R. S. Paul, and M. R. Wood. Feb. 14, 1956. Decl. Feb. 26, 1957. 67p. Contract W-31-109-Eng-52. $0.50(OTS).

An elementary description is given of a gamma-ray spectrometer and rotating-turret water-sampler system for monitoring the effluent water of a reactor for traces of fission products. Detailed instructions are given for the operation and maintenance of this equipment as specifically applied to Hanford. (auth)

NEUTRONS

13853 K-1309
Oak Ridge Gaseous Diffusion Plant, Tenn.
GENERAL APPLICATION OF A THEORY OF NEUTRON INTERACTION. H. F. Henry, J. R. Knight, and C. E. Newlon. Nov. 15, 1956. Decl. Feb. 25, 1957. 26p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

Experimental data for critical systems of interacting containers have been compared with simple 2-group calculations in which the interaction average fractional solid angle between components is determined as a function of the multiplication factor, k_1, of each container in the array. The results reveal that useful accuracy can be obtained

for all of the available experimental conditions, which include unreflected systems of cylinders and slabs containing high assay uranium solutions at hydrogen to U^{235} atomic ratios varying from 44 to 337. The minimum value of k_1 for which a pair of containers can be made critical with a fractional solid angle of 8% between them was found to be 0.94, and that for a fractional solid angle of 24% was 0.86. (auth)

13854 LA-1391
Los Alamos Scientific Lab., N. Mex.
MULTI-VELOCITY SERBER-WILSON NEUTRON DIFFUSION CALCULATIONS. Bengt Carlson. Mar. 24, 1952. Decl. Mar. 26, 1957. 35p. Contract W-7405-eng-36. $0.40(OTS)

The Serber-Wilson method is applied to specific examples of neutron diffusion, and the results are compared to those obtained using the integral theory method. (B.J.H.)

NUCLEAR PHYSICS

13855 K-343
Carbide and Carbon Chemicals Corp. K-25 Plant, Oak Ridge, Tenn.
CRITICAL MASS STUDIES. PART III. C. K. Beck, Dixon Callihan, J. W. Morfitt, and Raymond L. Murray. Apr. 19, 1949. Decl. Jan. 30, 1957. 77p. Contract [W-7405-eng-26]. $0.45(OTS).

The conditions under which U^{235} (93.4%), contained in aqueous solutions of UO_2F_2, becomes critical in cylindrical aluminum and stainless steel reactors were examined. (L.M.T.)

13856 KAPL-M-MLS-8
Knolls Atomic Power Lab., Schenectady, N. Y.
NOTE ON AGE CALCULATIONS FOR DIPHENYL–METAL MIXTURES. M. L. Storm and P. L. Hofmann. June 14, 1956. Decl. Mar. 14, 1957. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Calculations are made of the neutron age for a U^{235} fission source in diphenyl–zirconium and diphenyl–stainless steel mixtures. The semi-empirical procedure employed assumes the experimental age in water and employs average transport and slowing down cross sections calculated by R. W. Deutsch of KAPL. The ages so obtained at 600°F are compared with the results of earlier calculations at ANL which utilize Marshak's formula for mixtures of hydrogen with nonslowing-down heavy moderators. (auth)

NUCLEAR PROPERTIES

13857 NDA-Memo-15B-1
[Nuclear Development Associates, Inc., White Plains, N. Y.]
A PRELIMINARY DISCUSSION OF THE RATIO OF CAPTURE TO FISSION. (Preliminary Draft). V. F. Weisskopf. July 23, 1952. Decl. Mar. 18, 1957. 25p. $4.80(ph OTS); $2.70(mf OTS).

For energies below 500 ev, the capture-to-fission ratios of U^{235} and Pu^{239} may be obtained by measuring the peak values of σ_f and σ_t and using the relation $\alpha = (\sigma t/\sigma f)-1$. Two methods for measuring α at higher energies are discussed. The first method consists of wrapping samples in shields of different thicknesses and composition and irradiating them in a reactor. A mass spectrograph is then used to measure the amount of U^{236} or Pu^{240} produced. The second method involves the measurement of the danger coefficient for various samples in different reactors and different locations. A theoretical analysis of these procedures is given. (K.S.)

13858 WASH-188
Nuclear Cross Sections Advisory Group, AEC.
MINUTES OF MEETING OF AEC NUCLEAR CROSS SECTION ADVISORY GROUP, BERKELEY, CALIFORNIA, MAY 24 AND 25, 1955. C. W. Zabel. Oct. 1955. Decl. Mar. 5, 1957. 50p. $7.80(ph OTS); $3.30(mf OTS).

The cross-section measurement programs at the cooperating laboratories of the AEC are summarized. Specifications of the proposed boron glass secondary standards are included. (C.W.H.)

13859 Y-B23-22
Oak Ridge National Lab., Y-12 Area, Tenn.
RING TAMPING. Dixon Callihan and Frank Cronin. Nov. 5, 1952. Decl. Mar. 21, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

In a potential nuclear chain reacting system with two or more volumes of fissionable material surrounded by, but not separated by, a hydrogeneous neutron reflecting medium reliance is placed on neutron absorption by the H separating the components. The fissionable material used was U_3O_8 enriched to 5% in U^{235} mixed with glycerol tristearate for moderation. The H content of glycerol tristearate is 96% that of H_2O by volume and 111% by weight. The change in reactivity incurred by removal of the reflector material from between the components was determined. (F.S.)

NUCLEAR REACTORS

13860 AEP-101
Atomic Electric Project, St. Louis.
REACTOR PHYSICS STUDIES. G. E. Putnam and W. S. Hogan. Sept. 30, 1953. Decl. Feb. 7, 1957. 122p. $0.60 (OTS).

General reactor calculations are discussed with particular reference to their application in the theory of sodium-graphite reactors. Several methods for calculating K_∞ and K_{eff} are studied, together with related criticality theory dealing with long-term reactivity, the estimation of safety rod effects, and the production of isotopes. For purposes of rapid application of these theoretical methods to feasibility and design studies, attention is given in many instances to simplifying the calculating procedures in order that preliminary estimates can be obtained without recourse to elaborate and time-consuming analyses. (TRH)

13861 AMF-G-2495
American Machine and Foundry Co. General Engineering Lab., Greenwich, Conn.
DIMENSIONAL TOLERANCE REPORT OF FUEL ELEMENT DESIGN. A. Mirel, J. Bolton, and S. Cuthbert. Aug. 17, 1954. Decl. Mar. 19, 1957. 27p. (ER-6800). $7.80(ph OTS); $3.30(mf OTS).

In support of the fundamental aim of the Dow-Detroit Edison Nuclear Power Group to arrive at an optimum geometry for assembly of a fuel element, this report presents a technical analysis of the dimensional tolerances theoretically necessary to effect the assembly of three suggested designs; pancake, ribbed plate, and flat plate. The developed tolerances are very critical and are necessary to effect assembly of components having a minimum of clearance between the fuel and the coolant channel. (auth)

13862 AMF-GR-2-54(Del.)
American Machine and Foundry Co. General Engineering Lab., Greenwich, Conn.
PRELIMINARY STUDY OF SYSTEMS UTILIZING LOW ENRICHMENT FUELS. Reactor Research Report. S. Oestreicher and C. J. Brous. Oct. 1954. Decl. with deletions Apr. 12, 1957. 25p. (ER-6810(Del.)). $4.80(ph OTS); $2.70(mf OTS).

To-date practically all reactors have used either fully enriched or natural uranium fuels. In anticipation of the restriction of fuels to ones of low enrichment, this preliminary study was made to determine the effects on costs, operation, and design. It appears that research reactors, of the "swimming pool" and "water boiler" types can be built, using fuels of 20% enrichment, with only minor modifications in the core and auxiliary systems. The cost of 20% enrichment U is essentially the same as that for highly enriched U. The use of low enrichment fuel is justified solely by the fact that the fuel of low enrichment does not have weapons value. The cost of the basic components of a research reactor using low enrichment fuel will probably be not more than 20% higher than the cost of a similar flux level reactor using highly enriched fuel. Detailed calculations and designs are required to determine the effects of low enrichment on specific reactor systems. It is recommended that these calculations and designs be made in order to have available working drawings of research reactor systems suitable for both foreign and domestic use. Such a study can be divided into two principle phases: the nuclear physics and layout work costing approximately $18,000; and the design and detailing work costing approximately $59,200. In order to expedite completion of this study consultants reports and statements, and AEC Laboratory reports, memorandums, and statements have been quoted freely. (auth)

13863 ANL-4800(Del.)
Argonne National Lab., Lemont, Ill.
THERMAL NEUTRON FLUX DISTRIBUTION AND WILKINS EFFECT IN ONE-INCH FUEL RODS AND QUATREFOILS. J. L. Hyde and D. J. Pellarin. Aug. 1952. Decl. with deletions Feb. 10, 1957. 42p. Contract W-31-109-Eng-38. $7.80(ph OTS); $3.30(mf OTS).

A formula is derived for the thermal flux distribution in a single long 1-in. fuel rod in the $8\frac{1}{4}$-in. lattice of CP-2, which shows that simple diffusion theory is inadequate for accurate prediction of the distribution. A depression factor for a 1-in. rod in an $8\frac{1}{4}$-in. lattice is calculated. An approximate expression for the flux distribution in a long quatrefoil assembly containing four 1-in. rods spaced $1\frac{1}{4}$ in. between axes is derived, and the corresponding depression factor is calculated. The Wilkins effect for a single long rod, separated into 2 halves by a 0.575-in. Al spacer, is evaluated as the ratio of the average flux at the ends to the average flux at a location remote from the ends. The corresponding quantity for a quatrefoil, all 4 rods of which are separated by 0.575-in. spacers, is given. The flux distribution near the ends of level-ended rods, separated by Al spacers, as well as flat-ended rods, is also reported. (L.T.W.)

13864 ANL-5134
Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] JUNE 1, 1953 THROUGH AUGUST 31, 1953. Sept. 15, 1953. Decl. Mar. 28, 1957. 134p. Contract W-31-109-eng-38. $21.30(ph OTS); $6.90(mf OTS).

Progress is reported on the following studies: design and criticality calculations of the Argonne Power Breeder Reactor and the Central Station Water Reactor, water corrosion of Nb–U alloys, corrosion of welded Zr alloys, boiling heat transfer tests, speed of thermocouple response, control rod drive mechanisms, corrosion of Savannah River fuel slugs, rod drop experiments on ZPR-I, organic coolant studies, and temperatures produced by Na–Hg reactions. (L.T.W.)

13865 ANL-5186(Del.)
Argonne National Lab., Lemont, Ill.
REACTIVITY AS A FUNCTION OF IRRADIATION TIME IN THERMAL REACTORS. J. C. Carter and J. M. West. Dec. 1953. Decl. with deletions Feb. 11, 1957. 58p. Contract W-31-109-eng-38. $9.30(ph OTS); $3.60(mf OTS).

Equations governing the variation of U^{235}, U^{238}, Pu^{239}, Pu^{240}, and Pu^{241} were derived and their solutions plotted as a function of irradiation time. The initial U^{235} content of the U was varied from 0.5% to 2.0%. The range of conversion ratios was from 0.5 to 1.2. The irradiation was from 0 to 20,000 mwd/ton of fuel. Since a range of initial conversion ratios is associated with each value of enrichment, a solution results in a family of curves for each isotope, and, since the range of enrichments is large, the number of curves is quite large. Translation of the isotope curves to reactivity variation necessitates a calculation requiring a modest amount of time for a particular case but a prohibitive amount of time to cover the entire range of possible combinations of enrichment and initial conversion ratios. Reactivity variation as a function of irradiation time has been computed for a natural U reactor with an initial conversion ratio ranging from 0.7 to 1.2 and for 3 types of reactors in which there is considerable interest. Similar calculations for other reactors can be made by making use of the isotope curves and the calculation technique set forth. (auth)

13866 ANL-5208(Del.)
Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] SEPTEMBER 1, 1953 THROUGH NOVEMBER 30, 1953. Dec. 15, 1953. Decl. with deletions Apr. 16, 1957. 170p. Contract W-31-109-eng-38. $27.30(ph OTS); $8.10(mf OTS).

Power Breeder Reactor. The PBR, pilot plant, and EBR are described and compared. An optimum blanket cycle for PBR is analyzed with respect to costs, Pu inventory, Pu quality, and heat generation. A multigroup diffusion analysis of EBR and the pilot mockup was made. Experimental details and preliminary studies in connection with fuel elements, fuel media, reflector and structural materials are briefly reported. Thermal analyses of core, blanket, cooling cycle, and steam plant were carried out. Some characteristics of electromagnetic pumps and wetting of an electromagnetic flowmeter were also studied. Central Station Water Reactor. A 1,000 Mw reactor design and feasibility study are presented. Criticality and reactivity studies were made, and other reactor characteristics were investigated. Corrosion data on U–Nb and U–Nb–Zr alloys by water are given, and the results of burnout tests on Ni and stainless steel tubes at 1,000 and 2,000 psia are included. Reactor Physics. A Monte Carlo analysis of slant transmission and reflection of neutrons by water slabs of varying thickness was conducted, and a method for calculating coolant water activity for a boiling reactor was found. Corrosion studies. Corrosion inhibitors for type 347 stainless steel tested were: SiO_2, GeO_2, and ZnO. Instrumentation. The volt-ampere characteristic of Si, Se, and Ge diodes vs. neutron irradiation was studied. (T.R.H.)

13867 ANL-5244(Del. 2)
Argonne National Lab., Lemont, Ill.
SOLUBLE POISONS IN REACTOR CONTROL. C. R. Breden, W. S. Brown, and M. Sivetz. Nov. 1955. Decl. with deletions Mar. 18, 1957. 71p. Contract W-31-109-Eng-38. $10.80(ph OTS); $3.90(mf OTS).

Theoretical and experimental investigations of the use of soluble poisons to supplement mechanical control rods are summarized. Experimental evaluation of poisons of interest includes in-pile and out-of-pile tests simulating anticipated reactor operating conditions. Other phases of the investigation included methods of poison injection, re-

moval, and cleanup of poison-diluted reactor systems, as well as studies to evaluate possible application of soluble poison control in existing and proposed reactors. (auth)

13868 ANL-5260(Del. 2)
Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT [FOR] DECEMBER 1, 1953 THROUGH MARCH 30, 1954. Apr. 15, 1954. Decl. with deletions Apr. 8, 1957. 185p. Contract W-31-109-eng-38. $31.80(ph OTS); $9.30(mf OTS).

Boiling Experimental Reactor (BER). An H_2O-type reactor has been selected for an initial pilot boiling reactor. Information given here pertains to the D_2O design developed during the course of study. Plate-type and co-extruded tubular fuel element assemblies were considered for this reactor. Several cross section drawings of the proposed design are given. EBR-II. Several tank designs are proposed whereby the primary system, submerged in a Na-filled, double-walled tank, may be connected to the reactor vessel. A fuel element loading and unloading mechanism has been designed which incorporates a boom carried by a vertical column, and a moving head along the boom. Reactor Physics. Reactor physics calculations are reported on the feasibility of a pilot D_2O boiling reactor of 20 Mw heat output. The basic core design provides for fuel tube clusters, consisting of Zircaloy-clad fuel tubes containing slightly enriched U alloyed with Zr and Nb. Procedures for the determination of lattice constants are discussed, and the critical enrichment was determined for a core without burnup allowance. A similar analysis is made for a supercritical reactor. It was found that an enrichment was necessary for 10,000 Mwd/t exposure. The effects of moderator and fuel region D_2O density, and neutron and metal temperature on the reactivity of a boiling reactor are considered. Criticality changes resulting from small additions of H_2O and D_2O to the reactor were also evaluated. A number of activation cross sections were measured in various locations of the EBR. Tabular data of these cross sections referred to Au are presented for $Ni^{64}(n,\gamma)$, $Bi^{209}(n,\gamma)$, $Fe^{56}(n,p)$, $Zr^{96}(n,\gamma)$, $Zr^{94}(n,\gamma)$, $Fe^{58}(n,\gamma)$, and $Pb^{208}(n,\gamma)$ reactions. UNIVAC solutions of the multigroup diffusion equations in cylindrical geometry were obtained for a variety of specific problems related to fast reactors. Tables of cross section data and problem parameters are given. Fuel Elements. Several new techniques were tested for the fabrication of ball and wire fuel elements. Samples of stainless steel UO_2 fuel plates were corrosion tested in H_2O saturated with O_2 and H_2. The plates were not corroded during a 16-week exposure at 500, 600, and 680°F. Selected data are given for the water corrosion of Ni, Si, and Zr alloys of U in the unclad condition. Detailed information was obtained on the stages of corrosion penetration for a group of Zircaloy-clad U–Nb fuel plates in 500°F H_2O. Moderator, Reflector, and Structural Materials. $Zn(C_2H_3O_2)_2$ and ZnO were found to be effective corrosion inhibitors of 347 stainless steel and 2-S Al. An apparatus was constructed for thermal conductivity determininations of reactor materials up to 1000°F. The embrittling effects of irradiation, H_2, and temperature on the impact properties of materials were studied. The specimens tested included a variety of stainless steels, Ni–A, Ni alloys, Co alloys, and Zr. Reactor Coolant, Heat Transfer, and Power Generation. A new series of tests have been initiated on diphenyl as an organic moderator material, in order to determine the amount of carbonization on heat generating surfaces and the degree of decomposition as a function of neutron dosage. Tar deposits and an 11.1% decomposition was found in a closed loop test operated in the EBR for 73.5 hr at a power level of 1200 kw. Several thermal test loops for organic coolants are described. A liquid-metal level indicator has been developed which measures a change in the electrical inductance of a 60-cycle circuit, according to the amount of metal surrounding the test element. With a view of reducing high D_2O inventories for boiling D_2O reactors, an air-water atmospheric test facility was constructed in order to study separator configurations. A complete description of the apparatus is given, together with some representative test results and separator designs. A technique has been developed for the thermal analysis of complex fuel element designs under steady-state and transient conditions. An electrical, geometrical, analogue device is used, consisting of a resistive plotting surface of conducting paper, a d-c source, a resistive network coupling the plotting surface to the source, and a null-type voltage measuring device. Comparison of the analogue data with analytical results shows close agreement. A group of 16 fuel element configurations have been analyzed from the standpoint of maximum permissable temperatures and heat transfer characteristics, to determine the most feasible design for the PBR. The conclusions include other factors besides heat transfer considerations. A study was also made of the thermal conductivity characteristics of laminated fuel elements, where U particles and stagnant Na are arranged in layers within a containing structure. The condition of parallel heat flow is obtained when the laminations are made thin. Preliminary investigation of power plant cycle and reactor core designs using flashed water to produce steam indicates that a more detailed study is justified. Local heat transfer film coefficients of Na and NaK under conditions of high heat flux and non-circular channels will be measured in a heat transfer loop which is described. The construction of a 600-psi multiple channel boiling density test facility has been completed. A detailed description of the system and test section is presented. The facility will be used for the study of boiling H_2O phenomena at pressures near reactor operating conditions. The flow rate of naturally circulating boiling water was studied as a function of additives such as $HC_2H_3O_2$, Dow-Corning anti-foram, and caustic soda. Slow and rapid heat transients in boiling D_2O reactors were simulated in a water explusion test apparatus, which was used to obtain considerable data on the characteristics of a 7-tube fuel-element cluster. The removal of core decay heat in the event of a sudden reactor shutdown accompanied by a loss of coolant water will be assumed by the natural circulation of cooling air through the core. In connection with such an event several tests were made to determine the relationship between maximum metal temperature and the heat that can be dissipated to ambient air. Four- and five-ft channels, $\frac{3}{4} \times 4\frac{1}{2}$ in. were tested. Instrumentation and Control. A magnetic jack-type control rod mechanism was developed. The device is designed for operation in a pressure shell located in the reactor coolant. The performance of Na and H_2O seals was tested, together with H_2O-lubricated ball bearings mounted vertically under a thrust load. Irradiation Test Facilities. Thermal neutron flux measurements were made in the Argonne Water Loop at the MTR. (For preceding period see ANL-5208.) (K.S.)

13869 ANL-5461
Argonne National Lab., Lemont, Ill.
REACTOR ENGINEERING DIVISION QUARTERLY REPORT ON THE POWER REACTOR PROGRAM [FOR] JANUARY 1, 1955 THROUGH MARCH 31, 1955. Apr. 15, 1955. Decl. Jan. 17, 1957. 124p. Contract W-31-109-eng-38. $0.60(OTS).

Construction drawings and specifications have been completed for the building and power plant of the Experimental Boiling Water Reactor (EBWR). The design of the

pressure vessel has been evaluated for safety in case of internal explosion. The results of steam pipe rupture studies and tests of impurity carry-over in the boiling process are reported. The design and performance of a fuel assembly grappling tool are described. Design studies of control rods and control rod drive mechanisms are continuing. Tests are being made to observe the corrosion behavior of Zircaloy 2 coupled with types 304 and 347 stainless steels in static and dynamic high-temperature water. The results of stress analyses of EBWR type fuel plates and subassemblies are reported. Data are presented from corrosion and irradiation tests of high-uranium alloys, ThO_2–UO_2 bodies, and Al (Ni–plated or with Ni additions). Theoretical studies have been performed on the effect of the spatial distribution of voids on reactivity, and results are reported. The results of final shielding calculations for the EBWR are presented in graphical form. Calculations have been performed to determine the operating characteristics of D_2O-cooled and moderated, EBWR-type boiling reactors. Construction and pre-installation performance tests of the Experimental Breeder Reactor (EBR-II) Working Model are described. Low-frequency induction heating is being studied for the Na piping system. Convection heat transfer data are presented on eutectic NaK. Results of an EBR-II design study are reported, including a skeleton flow chart of the entire plant and a temperature-enthalpy diagram for the steam generator. A thermal analysis has been made of radiator-type fuel elements for the EBR-II, and results are presented. Results of irradiation tests of EBR-II pin-type fuel elements are summarized. Preliminary information from a set of UNIVAC multigroup calculations on the EBR-II is reported. (T.R.H.)

13870 ANL-5594(Del.)
Argonne National Lab., Lemont, Ill.
QUARTERLY PROGRESS REPORT ON REACTOR DEVELOPMENT 4000 PROGRAM. Stuart McLain, comp. Apr. 30, 1956. Decl. with deletions Mar. 14, 1957. 66p. Contract W-31-109-eng-38. $10.80(ph OTS); $3.90(mf OTS).

Development studies on the EBWR and operating data on the water and steam cycles of Borax III are reported. Operation of the EBR-II Working Model is discussed. Data are reported on thermal cycling of "fissium" alloys, U–Fe compound formation at Na-wetted surfaces of U and stainless steel, corrosive effects of diphenyl, corrosion of high-U alloys in 260°C water, corrosion of Al alloys in 290°C water, boiling corrosion of Al–Ni alloy, thoria–urania corrosion in 316°C degassed water, and B_4C solution in 215 and 260°C demineralized water. Fluoride volatility and pyrometallurgical separations processing studies are briefly discussed. (L.M.T.)

13871 ANP-58
Oak Ridge National Lab , Y-12 Area, Tenn.
THE MULTIGROUP METHOD AS USED BY THE ANP PHYSICS GROUP. D. K. Holmes. Feb. 15, 1951. Decl. Feb. 12, 1957. 38p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

The details are given of the method of obtaining the different equations which are used at the present time on the IBM by the ANP Physics Group for calculating quantities associated with reflected spherical reactors. The basic method and most of the detailed procedure is due to the Physics Division of KAPL. In particular, the method given here differs from the KAPL method in assuming that the average flux in a lethargy interval may be represented as arithmetical average of the values of the flux at the two lethargy limits of the interval. The KAPL method does not make the "linear" assumption, and assumes a relationship between the average and interval limit slowing down densities rather than fluxes. These differences result in considerable difference in the form of some of the final equations used for calculation, but the physical content is essentially the same. (auth)

13872 APAE-Memo-1
Alco Products, Inc., Schenectady, N. Y.
CRITICAL MASS CALCULATION FOR APPR-1 CRITICAL EXPERIMENT. J. G. Gallagher, H. W. Giesler, and W. R. Johnson. Sept. 23, 1955. Decl. Feb. 23, 1957. 10p. Contract [AT-(11-1)-318]. $1.80(ph OTS); $1.80 (mf OTS).

In new calculations to check previous calculations and to provide a tool for future work on APPR type reactors, the estimated critical mass was found to be larger than that predicted. The multiplication factor is smaller than predicted. (F.S.)

13873 APAE-Memo-17
Alco Products, Inc., Schenectady, N. Y.
HOT CHANNEL FACTORS AND POWER DISTRIBUTIONS FOR THE APPR-1. J. G. Gallagher. May 7, 1956. Decl. Feb. 22, 1957. 8p. Contract AT(11-1)-318. $1.80(ph OTS); $1.80(mf OTS).

13874 APAE-Memo-29
Alco Products, Inc., Schenectady, N. Y.
CURRENT STATUS OF APPR-1 FUEL ELEMENT METALLURGY. R. D. Robertson. July 23, 1956. Decl. Feb. 22, 1957. 16p. Contract AT(11-1)-318. $3.30(ph OTS); $2.40(mf OTS).

13875 APAE-Memo-57
Alco Products, Inc., Schenectady, N. Y.
THERMAL ANALYSIS OF APPR-1 CORE. III. C. H. Harvey. Sept. 17, 1956. Decl. Feb. 22, 1957. 18p. Contract AT(11-1)-318. $3.30(ph OTS); $2.40(mf OTS).

A coolant velocity distribution for maintaining uniform passage wall temperatures in the Army Package Power Reactor (APPR) is presented. The maximum possible wall temperature within the core can be reduced 41°F at the expense of a 4.7 foot increase in pressure drop in the primary system. (F.S.)

13876 APAE-Memo-61
Alco Products, Inc., Schenectady, N. Y.
RESULTS AND ANALYSIS OF THE APPR-1 ZERO POWER EXPERIMENTS. PART 1. H. W. Giesler, ed. Nov. 7, 1956. Decl. Feb. 22, 1957. 66p. Contract AT(11-1)-318. $10.80(ph OTS); $3.90(mf OTS).

The results of the initial experiments performed on the APPR-1 core at the Alco Critical Facility are given, and an interpretation of the experimental measurements, and a plan of operation for the APPR-1 are presented. An analysis of measurements made in the APPR-1 zero power experiments has indicated more than 15% excess reactivity in the cold clean core. Extrapolation of temperature coefficient data to 450°F predicts a temperature coefficient more negative than -2×10^{-4} $\Delta K/°F$. It is concluded that the APPR-1 may be operated safely on a five rod shim bank without a modification to the original B^{10} specification. (auth)

13877 APAE-Memo-68
Alco Products, Inc., Schenectady, N. Y.
EMERGENCY SHUTDOWN OF THE APPR-1 BY POISONING WITH BORON. J. L. Meem and G. W. Knighton. Nov. 26, 1956. Decl. Feb. 22, 1957. 4p. Contract AT(11-1)-318. $1.80(ph OTS); $1.80(mf OTS).

In the event the Army Package Power Reactor (APPR) is operated with completely arbitrary positioning of the rods, the reactor may be critical upon cooldown with two rods stuck. Provisions were made to shut down the reactor in

this case by injecting B in soluble form into the primary system. The equipment was sized for the worst case with three rods stuck out. (F.S.)

13878 APAE-Memo-79
Alco Products, Inc., Schenectady, N. Y.
NEUTRON FLUX SUPPRESSION IN APPR CONTROL RODS. W. R. Johnson and P. V. Oby. Jan. 10, 1957. Decl. Mar. 18, 1957. 24p. Contract AT(11-1)-318. $4.80(ph OTS); $2.70(mf OTS).

Haynes-25 alloy was chosen as the suppressor material for the APPR control rods despite its relatively low absorption cross section. Suppressor design, zero power measurements, thermal analysis, and flux and power analysis are discussed. (L.M.T.)

13879 APDA-105
Atomic Power Development Associates, Inc., Detroit.
FAST REACTOR CONTROL AND SAFETY MEETING [HELD AT DETROIT EDISON COMPANY ON] NOVEMBER 10-11, 1954. E. C. Branyan, ed.]. Mar. 11, 1957. 206p. $31.80(ph OTS); $9.30(mf OTS).

The report of the proceedings of a 2-day meeting on fast-reactor control and safety held in Detroit on Nov. 10-11, 1954 is given. The scope of the meeting was limited to the nuclear kinetics of fast reactors. (F.S.)

13880 BMI-916(Del.)
Battelle Memorial Inst., Columbus, Ohio.
TEMPERATURE DISTRIBUTIONS IN A REACTOR FUEL PLATE. IV. PARALLEL PLATES WITH NONUNIFORM HEATING. J. K. Long, R. G. Jung, R. F. Redmond, J. W. Chastain, and S. L. Fawcett. July 12, 1954. Decl. Mar. 5, 1957. 14p. Contract W-7405-Eng-92. $3.30(ph OTS); $2.40(mf OTS).

The MTR-type fuel plates of the proposed supercritical-water reactor are subject to a larger heat flux at the sides of the plates than at the center. The effect on fuel-plate temperature of this nonuniform heating and of the channel shape has been investigated. Local heat-transfer coefficients were measured on a large-scale model for the cases of uniform and nonuniform heating. The measured distribution of heat-transfer coefficients results in an estimated maximum wall temperature of about 1570°F. This is 180°F above a previous calculation based on equivalent circular-tube heat-transfer coefficients. (auth)

13881 BMI-1078
Battelle Memorial Inst., Columbus, Ohio.
SOLID-PHASE-REACTION STUDIES OF CONTROL-ROD MATERIALS. Arnold F. Gerds and Manley W. Mallett. Mar. 9, 1956. Decl. Mar. 12, 1957. 38p. Contract W-7405-eng-92. $0.40(OTS).

The solid-solid reactions between selected combinations of B-, Gd-, and Cd-bearing poisons (neutron absorbers) and several matrix metals were studied at 1750 and 2000°F. Matrix metals studied were Zircaloy 2, Ti, Ni, and stainless steel. Reaction specimens, consisting of poison powders compressed within containers of the matrix metals, were heated in graphite restraining dies under a dead load. Extent of reaction was determined by metallographic examinations and hardness traverses of cross sections of the specimen. Zirconium diboride appeared to be the least reactive boron-bearing poison tested in contact with Zircaloy 2 or stainless steel at 1750 and 2000°F and with nickel at 1750°F. Boron appeared to be less reactive than either B_4C or ZrB_2 with Ti. At 2000°F, GdO_2 with titanium or Zircaloy 2 reacted less than any other combination studied. Gadolinium carbide reacted only slightly with Zircaloy 2 at 2000°F. (auth)

13882 BNL-23
Brookhaven National Lab., Upton, N. Y.
REPORT ON THE ENGINEERING TESTS MADE IN CONJUNCTION WITH THE DESIGN OF THE BROOKHAVEN NUCLEAR REACTOR. W. E. Winsche, D. W. Bareis, J. Karp, and W. R. Page. Oct. 1, 1949. Decl. Sept. 21, 1956. 70p. $0.45(OTS).

The experimental work done with full-scale and small-scale mock-ups of sections of the Brookhaven Reactor cooling system indicates the split flow design to be feasible. A method was devised for calculating the air velocity and pressure pattern in the reactor slot for any combination of slot width, total air flow rate, reactor power level, and loading pattern. Consequently, a method for determining the required orifice pattern for any combination of the above conditions was developed. Tests performed on the reactor fuel assembly indicate that the heat transfer and air pressure drop characteristics may be correlated by the familiar empirical relations. Test results show that the fuel assembly will be physically stable under the conditions expected in the Brookhaven Reactor. (auth)

13883 BNL-113
Brookhaven National Lab., Upton, N. Y.
REACTOR SCIENCE AND ENGINEERING DEPARTMENT PROGRESS REPORT [FOR] JANUARY 1–JUNE 30, 1951. Decl. Feb. 14, 1957. 47p. $0.35(OTS).

Results are discussed of studies of a possible liquid-fuel reactor using a solution of U^{233} in liquid Bi with Be as the moderator. Data are shown of an investigation of the phase rule for the system U–Pb–Bi. Results of adhesion tension measurements on U_3O_8 in liquid Pb, Bi, or Sn are tabulated. Various steels have been tested at 500°C for static corrosion by U–Bi and other similar liquids. Various rare earths have been removed from U–Bi solutions by contacting with fused chlorides. The removal of Pa and U from ThF_4 has been investigated as part of a study of the suitability of ThF_4 as breeder blanket material. Results are discussed of corrosion tests of various alloys of Al, and of couplings between Al and other metals in a boiling mixture of BrF_3 and BrF_5. Results of measurements of γ-ray and neutron attenuations in Brookhaven concrete are reported. Data on the corrosion of 2S Al in air at 400 and 450°C are reported for exposures of 2500 hr. Various measurements on BNL reactor constants made during a start-up are discussed, most of them agreeing well with theoretical design predictions. An experiment allowing the reactor to reach low power at no cooling is described. The temperature of the metal went through several thermal oscillations before coming to equilibrium, and equations are derived to describe the observed effect. (D.E.B.)

13884 BNL-267
Brookhaven National Lab., Upton, N. Y.
QUARTERLY PROGRESS REPORT [FOR] AUGUST 16 – NOVEMBER 15, 1953. [Jan. 1954.] Decl. Apr. 8, 1957. 42p. $6.30(ph OTS); $3.00(mf OTS).

Progress is reported on the following studies: exponential experiments on water–slightly enriched U lattices, kinetics of a circulating-fuel reactor, inelastic neutron scattering cross section of natural Pb, nuclear temperature coefficients of U–graphite lattices, Liquid Metal Fuel Reactor component development, waste processing development, radiation effects on Brookhaven Reactor graphite, oxidation of Brookhaven Reactor graphite and analysis of products, self-diffusion of Fe in presence of radiation, formula for neutron streaming in steel slabs, γ-ray streaming through cylindrical voids in water, effects of impurities on streaming of neutrons through Fe, measurement of $(1 + \alpha)$ of U^{235} in the thermal region, and parameters for resonances in U^{238}. (B.J.H.)

13885 BNL-333
Brookhaven National Lab., Upton, N. Y.
QUARTERLY PROGRESS REPORT [FOR] NOVEMBER 16, 1954–FEBRUARY 15, 1955. Decl. Mar. 18, 1957. 56p. $0.40(OTS).

Studies of lattice constants for slightly enriched U rods in water have included measurements of buckling and the fast fission factor. Measurements were made of thermal neutron diffusion lengths of hydrogenous materials. The solution of specific problems in reactor theory, such as the kinetics of circulating fuel reactors, by means of electronic computers is described. Studies are reported on the proposed modification of the Brookhaven Reactor to permit operation at higher flux levels. Methods of power regulation for the Liquid Metal Fuel Reactor (LMFR) have been investigated. The results of salt–metal equilibration experiments have been reappraised to find a probable value for the separation factor and to determine the reasons for the large scatter of the data. A proposed flowsheet for an LMFR fuel processing system is shown. The heat released by fission products in LMFR fuel has been estimated. The corrosive effects of fused LiCl–KCl salt on container materials have been studied under static and dynamic conditions. The results of phase studies on Th–Bi–Pb, Th–Bi, and U–Bi systems are reported. The production of Th–Bi dispersions from Th powder, particle formation during thermal cycling of Th–Bi dispersions, and the diffusion of Pa in Th–Bi dispersions are being studied. The performance of a number of liquid-metal natural convection loops is summarized. Data are presented on the solubility of Cr in Bi. The rate of solution and precipitation of Fe from Bi has been measured in Croloy and pure Fe crucibles, and data are presented. The reactions of graphite with U in Bi–U solutions are being studied. Thermal and in-pile graphite oxidation data are summarized. Work on the transmission of neutron and γ rays through holes and ducts in water shields was continued. (T.R.H.)

13886 BNL-2514
Brookhaven National Lab., Upton, N. Y.
LMFR PROGRESS LETTER FOR AUGUST 1955. F. T. Miles. Sept. 26, 1955. Decl. Mar. 18, 1957. 3p. $1.80 (ph OTS); $1.80(mf OTS).

13887 CE-1143
[Chicago. Univ.] Metallurgical Lab.
MAXIMUM UTILIZATION OF P-9 [D_2O] IN HOMOGENEOUS SLURRY PILES. J. R. Huffman. Dec. 15, 1943. Decl. Feb. 18, 1957. 38p. Contract [W-7401-eng-37]. (A-1630). $7.80(ph OTS); $3.30(mf OTS).

13888 CE-2554
[Chicago. Univ. Metallurgical Lab.]
TEMPERATURE RELATIONSHIP BETWEEN THE CENTER OF AN X TYPE SLUG AND THE END OF ITS JACKET. Final Report on a Phase of Problem No. 390-ML-51-S. R. N. Lyon. Dec. 21, 1944. Decl. Feb. 15, 1957. 8p. Contract W-7401-eng-37. $1.80(ph OTS); $1.80(mf OTS).

Calculations were made to determine the relationship between a Clinton slug center temperature and the temperature at the center of its can end. It is recommended that the can end temperature of a Clinton slug not be used as an indication of the temperature at the center of the slug. (auth)

13889 CERD-S1C-106
Combustion Engineering, Inc. Reactor Development Div., Windsor, Conn.
PRIMARY SHIELD MATERIALS STUDY. G. P. Lewis. Sept. 28, 1956. Decl. Apr. 4, 1957. 30p. Contract AT(30-3)-198. (S1C/MD/56-102). $0.35(OTS).

Primary shields constructed of Pb, mild steel, and Pb-mild steel combinations have been proposed for S1C. Calculations indicate 25,000 pounds of lead can be saved through the use of borated H_2O if the associated corrosion problems can be solved. An investigation was initiated to determine the compatibility of these shield materials with borated solutions. Results of 1000 hour corrosion studies, boron compound solubilities, and boron compound toxicity studies are reported. (D.E.B.)

13890 CF-49-9-114
Oak Ridge National Lab., Tenn.
PRODUCTION OF GASEOUS FISSION PRODUCTS IN HOMOGENEOUS REACTOR. H. E. Goeller. Aug. 25, 1949. Decl. Apr. 8, 1957. 23p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

Calculations were made to determine the activity and volumes of the gaseous fission products, Br, Kr, I and Xe, produced in the homogeneous reactor and adsorbed in a set of off-gas adsorbers. The maximum reactor power level was assumed to be 200 kw. Calculations were made for degassing rates of two and twenty percent of the reactor recycle rate of 30 gpm, resulting in degassing cycles of 27 and 2.7 minutes respectively. (auth)

13891 CF-51-8-137
Oak Ridge School of Reactor Technology, Tenn.
POWER PRODUCER. Reactor Design and Feasibility Problem. B. W. O. Dickinson, K. A. Hub, G. P. Letz, T. A. Redfield, J. N. Renaker, R. A. Woll, and R. J. Beeley. Aug. 6, 1951. Decl. Mar. 12, 1957. 64p. $10.80 (ph OTS); $3.90(mf OTS).

From criticality calculations a homogeneous reactor which is a 27-ft-diameter sphere containing D_2O-UO_3 slurry at 25°C and 1000 psi was selected. The reactor designed produces only power; the Pu produced in the reaction serves as fuel. All mechanical, chemical, nuclear, and electrical engineering calculations are given, and costs are estimated. (auth)

13892 CF-51-9-155(Del.)
Oak Ridge National Lab., Tenn.
PRELIMINARY DESIGN CONSIDERATIONS OF A SUPERHEATER UTILIZING THE RECOMBINATION ENERGY OF THE D_2-O_2 GAS MIXTURE PRODUCED IN THE FULL-SCALE HOMOGENEOUS REACTOR. J. T. Agnew. Sept. 17, 1951. Decl. with deletions Feb. 13, 1957. 36p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

A superheater will be designed on the basis of a shell pressure of 20 psia. The D_2O will condense at 228°F. Heat will be extracted from D_2O vapor until its temperature is 500°F. The remaining energy in the vapor and the condensation energy will then be extracted in a feed water heater section. (F.S.)

13893 CF-52-5-226
Oak Ridge National Lab., Tenn.
SLURRY AND BOILING REACTOR RESEARCH. R. N. Lyon. May 12, 1952. Decl. Apr. 2, 1957. 32p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

The physical properties of reactor slurries are briefly discussed. Boiling reactor studies include work on power density, the expected volume of boiling solutions, volume boiling of aqueous solutions, the promotion of boiling by fission in superheated U solutions, boiling reactor kinetics, and a small scale boiling reactor experiment. (B.J.H.)

13894 CF-53-1-294
Oak Ridge National Lab., Tenn.
PREDICTION OF CRITICALITY BEHAVIOR OF A SLURRY UPON SETTLING. P. R. Kasten. Jan. 26,

1953. Decl. Apr. 2, 1957. 20p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

An investigation has been made of the criticality behavior of a slurry reactor as a function of settling of the fuel particles. For a given reactor with an initial H/U^{235} ratio above a certain value, a super-critical condition can be attained as a result of slowing down of the stirrer motor. Complete stopping of the stirrer motor should lead to a less dangerous situation. The slope of the U^{235} critical mass (C.M.) vs H/U^{235} ratio curve appears important in determining the criticality behavior upon fuel settling, a super-critical condition being attainable when $d(C.M.)/d\ H/U^{235}$ is positive, zero, or slightly negative. For more negative values of $d(C.M.)/(d\ H/U^{235})$ slurry settling will result in a sub-critical assembly. (auth)

13895 CF-53-2-115
Oak Ridge National Lab., Tenn.
PROPOSAL FOR OPERATION OF LITR AT HIGHER POWER LEVELS. T. E. Cole. Feb. 13, 1953. Decl. Apr. 2, 1957. 23p. Contract [W-7405-eng-26]. $4.80 (ph OTS); $2.70(mf OTS).

A proposal is presented for the experimental operation of the LITR at power levels sufficiently high and for periods of time sufficiently long to demonstrate the feasibility and safety of operation at 3000 kw, the maximum power to be no greater than 3000 kw and the duration of any given run at a power greater than 1500 kw to be no longer than six days. A summary of collected information, calculations, estimations, and opinions as to the feasibility of higher power operation, and a cost estimate are appended. (C.H.)

13896 CF-53-4-57
Oak Ridge National Lab., Tenn.
COMMENTS ON VAPOR SEPARATORS IN BOILING REACTORS. P. C. Zmola and R. N. Lyon. Apr. 1, 1953. Decl. Feb. 15, 1957. 9p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

Lack of knowledge of operating characteristics of vapor separators prevents the determination of operating limits of a boiling reactor; thus, it is concluded that 25% of the effort in the boiling reactor program should be devoted to vapor separators. The information will be applicable to any reactor system which uses phase changes for power removal and will complement work on gas separators in circulating homogeneous systems. (T.R.H.)

13897 CF-53-5-190
Oak Ridge National Lab., Tenn.
SOLID FUEL PLATES OF ALUMINUM AND U_3O_8. E. S. Bomar. May 22, 1953. Decl. Apr. 2, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Uranium(IV) oxide was found to be unstable in pressed powder cores of plates processed by the MTR schedule. However, cores containing up to 47.2% by weight of U_3O_8 were successfully clad. The first pressed powder cores rolled were found subject to surface and edge cracking. The severity of cracking increased with the percentage of U_3O_8 present. This type of defect was almost completely eliminated by increasing the width of the side members of the picture frame. Presintering of cores also appeared to improve their rolling properties. Inconsistencies were found among the results of three methods of checking for homogeneity of U_3O_8 distribution. X-ray pictures and density determinations indicated a uniform distribution existed throughout the rolled plates; while chemical analyses showed a deviation as high as 8% for uranium concentration along one plate. Segregation difficulties have been encountered with attempts to cast Al–U alloys containing more than 20% by weight of U. There are applications for MTR type fuel elements in which U investments in excess of this value are desirable. As a possible means of circumventing this trouble, the techniques of powder metallurgy have been applied to the problem. (auth)

13898 CF-53-7-131
Oak Ridge National Lab., Tenn.
REPORT ON THIRD AND FOURTH ARE CLEANING COMMITTEE MEETINGS. G. M. Adamson. July 15, 1953. Decl. July 5, 1957. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Problems are discussed pertinent to the post-operative cleaning and decontamination of the Aircraft Reactor Experiment (ARE). (T.W.S.)

13899 CF-53-9-27(Del. 2)
Oak Ridge National Lab., Tenn.
REVIEW OF ORNL PROGRAM FOR THE DEVELOPMENT OF ECONOMICAL NUCLEAR POWER. J. A. Swartout. Sept. 11, 1953. Decl. with deletions Mar. 12, 1957. 12p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Brief statements by laboratory personnel are presented delineating the status of the ORNL Power Reactor Program as of Sept. 4, 1953. (T.W.S.)

13900 CF-53-9-94
Oak Ridge National Lab., Tenn.
FEASIBILITY OF A FLUIDIZED THORIUM OXIDE BLANKET. P. R. Crowley and A. S. Kitzes. Sept. 2, 1953. Decl. Feb. 15, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The feasibility of a fluidized bed blanket for the Intermediate Scale Homogeneous Reactor is studied. The specifications which must be met by the blanket are discussed. Data are given which allow selection of a thorium oxide density and fluidized bed porosity for any concentration and minimum fractional height to which the bed may settle. Diagrams are shown of the reactor using two types of fluidized beds. (B.J.H.)

13901 CF-53-9-179
Oak Ridge National Lab., Tenn.
SOLID FUEL PLATES OF ALUMINUM AND U_3O_8. E. S. Bomar. Sept. 7, 1953. Decl. Feb. 13, 1957. 13p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

X-ray pictures of solid fuel plates of Al and U_3O_8 indicate uniform blend of the cores. Density measurements, using water displacements, also showed uniform U_3O_8 distribution, but chemical analysis by potentiometric titration failed to show uniform U_3O_8 distribution. Samples indicate core thickness variations could cause observed deviations but would have no effect on the total U content which can not be checked by density measurements to closer than 10% when the degree of U_3O_8-to-UO_2 conversion is unknown. (F.S.)

13902 CF-53-10-22
Oak Ridge School of Reactor Technology, Tenn.
10-MEGAWATT AQUEOUS HOMOGENEOUS CIRCULATING SOLUTION REACTOR FOR PRODUCING ELECTRICAL POWER IN REMOTE LOCATIONS. D. W. Montgomery, W. J. Dodson, F. F. Kaiser, W. K. Luckow, and T. J. Pashos. Aug. 1953. Decl. Feb. 23, 1957. 167p. $1.00 (OTS).

The preliminary design of a 10 Mw power plant utilizing a circulating fuel-type homogeneous reactor as the primary heat source is presented. The power plant is designed to satisfy the military requirements of 1000 kw of net electrical output and 3800 kw of space heat. The "package" unit, containing the heat exchanger and the core, is a removable element 17.5 ft high and 6.67 ft in diameter. Provisions for detachment of both this unit and the main pump internals have been included. Uranyl-sulfate (UO_2SO_4 with 93.5% enrichment of U^{235}) is used as

the fuel and D_2O is used as the moderator. The critical concentration is 4.9 g/l for a clean reactor. The entire volume of the system, including core and external hold-up, is 1555 l. This volume and concentration result in an initial uranium loading of 7.62 kg. New fuel is to be added at short intervals after startup. The entire fuel loading is to be processed after five years of operation. (auth)

13903 CF-53-10-25
Oak Ridge School of Reactor Technology, Tenn.
STUDY OF A FUSED SALT BREEDER REACTOR FOR POWER PRODUCTION. D. B. Wehmeyer, J. A. Bara, Jr., D. J. Cockeram, R. B. Donworth, L. B. Holland, R. S. Hunter, P. J. Mraz, W. Park, and W. L. Webb. Sept. 1953. Decl. July 5, 1957. 137p. $22.80(ph OTS); $7.20(mf OTS).

The possibility of using a fused salt breeder reactor as a heat source for a central station power plant is investigated. Calculations are made for a homogeneous $Th-U^{233}$ breeding system in a fused $LiF-BeF_2$ salt. Analysis of associated chemical and engineering problems is presented. Economic analysis shows a cost of 6.6 mills/kwh for 235 Mw of net electrical output. A summary is included of the research needed to determine feasibility. (auth)

13904 CF-54-2-200
Oak Ridge National Lab., Tenn.
CONTROL OF XENON POISONING BY STRIPPING WITH GAS. R. E. Aven. Feb. 19, 1954. Decl. Feb. 14, 1957. 17p. Contract [W-7405-eng-26]. $0.30(OTS).

The amount of decomposition gas needed to remove Xe^{135} from the fuel system of homogeneous reactors using aqueous uranyl sulfate fuels was calculated. (C.H.)

13905 CF-54-8-53(Rev. 3)
Oak Ridge National Lab., Tenn.
HRT DESIGN DATA. Issue No. 3. F. C. Zapp and R. E. Aven. Apr. 19, 1956. Decl. Mar. 12, 1957. 21p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Design data sheets are given for the Homogeneous Reactor Test (HRT). The data are for a two-region power producer having an over-all gross design power rating of 5220 kw. This memorandum covers the major equipment in both the fuel and blanket loops. Detail design data on other equipment associated with the HRT will appear in the HRT Design Manual. (auth)

13906 CF-54-8-240
Oak Ridge School of Reactor Technology, Tenn.
BOILING HOMOGENEOUS REACTOR FOR POWER AND U^{233} PRODUCTION. Reactor Design and Feasibility Problem. H. R. Zeitlin, H. W. Bertini, W. F. Bourgeois, J. N Calvin, J. R. Engel, G. H. Farbman, O. R. Meyer, and D. P. Ross. Aug. 1954. Decl. Mar. 14, 1957. 172p. $1.10(OTS).

Design parameters are reported for a central-station single-region boiling homogeneous reactor fueled with a slurry of UO_2-ThO_2 in D_2O and having a thermal capacity of 500 Mw. Analyses of associated problems of control, power removal, and chemical processing systems are included. (T.W.S.)

13907 CF-55-1-183
Oak Ridge National Lab., Tenn.
MINUTES OF HRP GROUP LEADERS MEETING DECEMBER 16, 1954. H. F. McDuffie. Jan. 31, 1955. Decl. Feb. 15, 1957. 7p. Contract [W-7405-eng-26]. $1.80 (ph OTS); $1.80(mf OTS).

The status of design and fabrication of the Homogeneous Reactor Test and a review of studies for the Thorium Breeder Reactor are reported. General fields of effort for the Metallurgy Group are outlined. (F.S.)

13908 CF-55-2-178
Oak Ridge National Lab., Tenn.
EXPERIMENTAL PHYSICS PROGRAM FOR THE HRT. Sidney Visner. Feb. 28, 1955. Decl. Apr. 2, 1957. 23p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

A brief outline is given of the purposes and methods of the physics experiments, having to do with reactor statics, kinetics, and stability, which should be performed on the HRT. (M.H.R.)

13909 CF-55-7-141
Oak Ridge National Lab., Tenn.
HEATING IN HRT COLD TRAPS FROM RADIOACTIVE FISSION PRODUCT GASES. R. Van Winkle. July 28, 1955. Decl. Feb. 16, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Estimates are given of heat load from radioactive decay of krypton and xenon for various assumed flows of excess oxygen from the core system. The heat load appears to be relatively insensitive to changes in oxygen flow in the range of 125 to 500 cc/min. The estimated radioactive energy that must be dissipated by the cold trap appears to be about 3000 Btu/hr if the reactor operates at 10 Mw. (auth)

13910 CF-56-3-42 and Add.
Oak Ridge National Lab., Tenn.
HRT OXYGEN ADDITION SYSTEM. F. C. Zapp. Mar. 2, 1956. Decl. Mar. 14, 1957. Addendum: Mar. 21, 1956. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80 (mf OTS).

The addition of oxygen to the inlet or low pressure side of the Homogeneous Reactor Test fuel pumps is considered; oxygen addition at low reactor power levels is known to prevent uranyl ion reduction within the fuel solution and to prevent excessive corrosion of the stainless steel system. Details of addition system components and operating procedures are included. (T.W.S.)

13911 CF-56-4-29
Oak Ridge National Lab., Tenn.
URANIUM INVESTMENT IN A CIRCULATING FLUORIDE POWER-PRODUCING REACTOR. W. K. Ergen. Apr. 4, 1956. Decl. Mar. 26, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Combination of critical radius and critical concentration (in grams U^{235}/cm^2) are given for a bare reactor having UF_4 dissolved in $NaZrF_5$. (L.M.T.)

13912 CF-56-4-66
Oak Ridge National Lab., Tenn.
HRT FUEL CONCENTRATION AS A FUNCTION OF POISON LEVEL. C. W. Nestor, Jr., and P. R. Kasten. Apr. 11, 1956. Decl. Mar. 18, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Placing corrosion specimens within the core region will increase the fuel concentration required for criticality, due to the neutron poisoning effect of the specimens and their holders. Calculations based upon a preliminary coupon holder design indicate that the poisoning effect will be less than about 11% poisons. This maximum poisoning would increase the core critical concentration by ~23% at 280°C (from ~7.1 gm 25/kg D_2O to ~8.7 gm 25/kg D_2O). (auth)

13913 CF-56-8-28
Oak Ridge National Lab., Tenn.
THE CURRICULUM OF THE OAK RIDGE SCHOOL OF REACTOR TECHNOLOGY. L. G. Alexander. Aug. 6, 1956. 34p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mc OTS).

The ORSORT curriculum, as it has evolved in six years under the direction of Dr. F. C. Vonder Lage, comprises courses in Reactor Analysis, Engineering, Materials, Chemical Technology, Shielding and Health Physics, Control and Instrumentation, Systems, Component Design and Special Topics, Reactor Physics Laboratory, and Project (group theses). The lecture courses are discussed from the standpoint of objectives, content, administration, and texts. Detailed outlines are given. (auth)

13914 CF-56-8-128
Oak Ridge National Lab., Tenn.
THE APPLICATION OF LOW-ENRICHMENT URANIUM DIOXIDE TO ALUMINUM PLATE-TYPE FUEL ELEMENTS. R. C. Waugh and J. E. Cunningham. Aug. 20, 1956. Decl. Apr. 1, 1957. 17p. Contract [W-7405-eng-26]. $0.30(OTS).

The fabrication of fuel plates for the Geneva Reactor is reviewed. These plates, 20% enriched UO_2–Al powder, Al clad, were fabricated by hot pressing and rolling. The problem of chemical reaction of the meat components is discussed, and the element performance is predicted from test results. (D.E.B.)

13915 CF-3290
[Chicago. Univ. Metallurgical Lab.]
SKETCH OF A GAS TURBINE POWER PLANT. Gale Young. Aug. 10, 1945. Decl. Mar. 12, 1957. 14p. Contract W-7401-eng-37. $3.30(ph OTS); $2.40(mf OTS).

Some calculations are made for a Be metal pile cooled by He operating in a gas turbine cycle to give mechanical power output. An efficiency of about 20% is indicated. (auth)

13916 COO-208(Del.)
Chicago Operations Office, AEC.
AN ECONOMIC STUDY OF LIGHT AND HEAVY WATER REACTORS. Alan M. Hallene. Apr. 15, 1953. Decl. with deletions Mar. 7, 1957. 31p. (RD-346(Del.)). $6.30(ph OTS); $3.00(mf OTS).

Two different methods were used to evaluate light and heavy water reactors. The first evaluations considered power to be the primary product and Pu to be a salable by-product. The second group of evaluations considered power to be the only salable product. Curves are given which show the concentration of constituents as a function of exposure, cost of Pu vs. exposure, and investment vs. power cost. The evaluation indicates that the H_2O system is more favorable economically. (B.J.H.)

13917 CRD-R-21(Del. 2)
California Research and Development Co., Livermore, Calif.
NOTES ON THE DEVELOPMENT OF POWER REACTORS. H. W. Crandall and W. K. Davis. Dec. 22, 1952. Decl. with deletions Feb. 26, 1957. 28p. (LWS-24680(Del. 2)). $4.80(ph OTS); $2.70(mf OTS).

An examination is undertaken of the comparative merits of thermal, intermediate, and fast reactors, using U^{233} and Pu^{239} as fuel material and Th and U as fertile material, for realizing economically competitive power production without including credit for by-product fissionable materials. Attention is given to design factors influenced by fuel inventories, economical fuel processing, invested capital costs, fuel cycles, reactor location, and neutron economy. It is concluded that the Pu fast breeder offers the most promising compromise for the stated conditions. (D.E.B.)

13918 DP-147(Del.)
Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.
THE NUCLEAR TEST GAUGE. Thomas F. Parkinson, Malcolm H. Goosey, William M. Heston, and Herbert J. Hollberg. June 1956. Decl. with deletions May 8, 1957. 41p. Contract AT(07-2)-1. $7.80(ph OTS); $3.30(mf OTS).

The "Nuclear Test Gauge", a slightly subcritical assembly for quality control of reactor components, was designed, constructed, and placed in operation. Calibration experiments demonstrated that the sensitivity of the unit is adequate, and that the speed of testing is increased by a factor of ten over that of a critical test pile. (auth)

13919 HW-31385(Del.)
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
RDS-10 AND 11 — A PROPOSED OVERALL WATER SYSTEM FOR A DUAL-PURPOSE REACTOR. W. L. Pearl, J. C. Whipple, and K. L. Harms. Apr. 19, 1954. Decl. with deletions May 8, 1957. 51p. Contract W-31-109-Eng-52. $9.30(ph OTS); $3.60(mf OTS).

Development effort sponsored by RDS-D-10 and RDS-D-11 is being applied to systems development for the advancement of the state of the science of reactor design and technology. These schemes include water cooled, plutonium producing reactors utilizing high temperature recirculation cooling systems. Electrical power may be produced as a by-product from the waste energy. An over-all water flow diagram and water treatment plan form an integral part of these system studies. It is recognized that any over-all water system is subject to changes, dependent upon the outcome of further development studies of the design and economics of the various components, e.g., reactor, pumping plant, electrical power plant, etc. An engineering basis is formulated for the over-all water system required which could be used in conjunction with a liquid-phase reactor operating with a high temperature recirculating cooling system, producing plutonium and producing by-product electrical power from the waste energy (DPR). It is a secondary purpose to recognize the additional development work required to convert from liquid-phase operation to in-pile boiling operation. (auth)

13920 HW-32988
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
PROCESS SPECIFICATIONS—REACTOR COOLING WATER TREATMENT. Oct. 1, 1954. Decl. Apr. 4, 1957. 16p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40 (mf OTS).

13921 HW-38780(Rev.)
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
STATISTICAL ANALYSIS OF RUN TO RUPTURE TESTS INVOLVING MORE THAN TWO METAL TYPES. J. L. Jaech. Nov. 8, 1955. Decl. Feb. 26, 1957. 13p. Contract W-31-109-Eng-52. $0.25(OTS).

Two specific problems are considered for the case when more than two metal types are to be compared by means of a run-to-rupture-type test. A test is first made to determine whether or not the m metal types differ significantly among themselves as far as rupture is concerned. If overall differences are found to exist, it is desirable to classify the m types into subgroups such that within any subgroup the metal types do not differ significantly. Statistical methods for analyzing both cases are given. (B.J.H.)

13922 HW-43557
Hanford Atomic Products Operations, Richland, Wash.
METALLURGICAL EXAMINATION OF H-LOOP FUEL ELEMENTS RESULTING FROM RUPTURE INCIDENT. G. R. Mallett. Aug. 13, 1956. Decl. Mar. 13, 1957. 16p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

The first and third slugs downstream from a ruptured slug and a piece of the can wall from the ruptured slug were examined, and the extent and type of Al corrosion were determined. It was found that the ruptured slug and the third one became severely warped. The rupture acted as a water heater causing inter-granular corrosion and excessive heat generation. The over-heated water caused Al intergranular corrosion. The indications are that the higher the water coolant temperature the less warp can be tolerated. (F.S.)

13923 IDO-16347

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

EFFECT ON UNEVEN FUEL BURNOUT IN MTR ON MAXIMUM HEAT FLUX. J. H. Rainwater and V. A. Walker. Feb. 24, 1956. 12p. Contract AT(10-1)-205. (PTR-49). $3.30(ph OTS); $2.40(mf OTS).

The present mode of fuel loading in the MTR wherein elements with about 20% burnout are loaded in the central part of the core results in a nonuniform fuel distribution in these elements. Calculations have been made in an attempt to assess the magnitude of this nonuniformity on the predicted maximum heat flux. Results of the calculations show that the predicted heat flux is about 10% less when nonuniformity of fuel is taken into account if the hot spot is in an element which has considerable burnout. Results of calculations on Charge 58A are presented. Since some evidence exists that a very small amount of boiling may have occurred in Charge 64B and the calculated heat flux for this charge by the normal method was less than other loadings, the calculation of heat flux using the nonuniform fuel distribution will be made for all charges since raising the power to 40 Mw. A later report will discuss observations made in Charge 64B and show heat flux calculations for all charges since raising the power to 40 Mw. (auth)

13924 IDO-16350

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

DETERMINATION OF PERMISSIBLE MTR POWER FOLLOWING A JUNIOR SCRAM. J. H. Rainwater and V. A. Walker. Mar. 29, 1956. 24p. Contract AT(10-1)-205. (PTR-62). $4.80(ph OTS); $2.70(mf OTS).

The results of a flux run in which the Junior Scram condition was simulated permitted a test to determine the maximum permissible power with Be shim rods 6 and 7 fully inserted. This test showed that a reactor power of 33 Mw is permissible during the recovery from a Junior Scram if reactor operation had been without evident abnormality prior to the Junior Scram and the average fuel shim rod position is more than 20 inches during the recovery period. (auth)

13925 IDO-16351

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

PROPOSED FUEL LOADINGS FOR 600 MWD OPERATION OF THE MTR. J. H. Rainwater and V. A. Walker. Apr. 17, 1956. 8p. Contract AT(10-1)-205. (PTR-67). $1.80 (ph OTS); $1.80(mf OTS).

Since the reactor power was raised to 40 Mw, a mid-cycle shutdown has been used because it was felt to be necessary to have critical shim rod positions of at least 19 inches. A proposed loading system is presented herein which will run for at least 600 Mwd with a maximum heat flux no greater than the heat fluxes used in previous cycles. Since operating experience has shown that less than half of the cycles required mid-cycle reloading due to experiment scrams, a net gain in operating time will be made with this type loading with no sacrifice in fuel economy. The loading described uses two new shim rods each cycle rather than the one presently used. This may be cut back to one shim per cycle by loading Position 23 with a 200 gram element rather than the 170 gram element shown. If the proposed loading had been in use since raising power to 40 Mw, a saving of 89 hours operating time would have been made. This time is worth about $62,000. The additional shim rod cost would have been about $15,000 resulting in a net economic gain. (auth)

13926 IDO-16354

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

EXAMINATION OF BORON CONTENT IN ETR 1-4 FUEL ELEMENT BY MEANS OF FUEL SCAN AND TRANSMISSION MEASUREMENTS. W. C. Francis. May 25, 1956. 9p. Contract AT(10-1)-205. (PTR-83). $1.80(ph OTS); $1.80(mf OTS).

In an effort to determine its B content and distribution, the ETR 1-4 fuel element was scanned in the standard MTR fuel assay equipment, and then subjected to transmission measurements in the crystal spectrometer. A conventional MTR fuel element was used for comparison purposes. Neither test gave sufficient sensitivity for a precise determination of B content although readings were within 30% of stated B content. The test procedures did not disclose any serious nonuniformities in B distribution. The need for a more adequate method, however, is indicated. (auth)

13927 IDO-16357

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho

NUCLEAR CALCULATIONS FOR THE MTR WITH GENEVA FUEL. H. L. McMurry, R. Haldin, and L. L. Marsden. July 25, 1956. 8p. Contract AT(10-1)-205. (PTR-98). $1.80(ph OTS); $1.80(mf OTS).

Excess reactivities of the MTR fuel with 20% enriched U (Geneva type fuel) and regular MTR (highly enriched) fuel have been calculated. A resonance escape probability of p = .93 was found to give good agreement with the critical data on the Geneva reactor and was used in the calculations. It appears that the low enriched fuel will require about 40 g more of U^{235} per fuel assembly to give a charge life equivalent to that obtainable from highly enriched fuel. (auth)

13928 KAPL-436(Suppl. 1)

Knolls Atomic Power Lab., Schenectady, N. Y.

FEASIBILITY REPORT FOR THE KAPL THERMAL TEST REACTOR. Supplement No. 1. H. B. Stewart, F. G. LaViolette, C. L. McClelland, G. B. Gavin, and T. M. Snyder. Apr. 3, 1951. Decl. Apr. 4, 1957. 24p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

The TTR has been built at the KAPL Sacandaga Site and operated critically for a number of hours. The reactivity sensitivity of the central sample hole, the value of the safety and control rods, and the temperature coefficient of the reactor were all found to be essentially as predicted. A proposed design is given for housing the TTR at the Knolls site. A study of the higher wattage operation of the TTR indicates that the hazards involved in any possible accident are not increased by using higher power. (M.H.R.)

13929 KAPL-988

Knolls Atomic Power Lab., Schenectady, N. Y.

CONTROL AND ENDURANCE EVALUATION OF A POWER REACTOR BY MEANS OF A CRITICAL EXPERIMENT. L. Tonks. Sept. 18, 1953. Decl. Mar. 27, 1957. 20p. Contract W-31-109-Eng-52. $0.25(OTS).

Critical experiments are useful for determining the safety margin and endurance and for calculating control characteristics of a power reactor, but it is not self-evident how experimental results should be interpreted to obtain these characteristics, and there has been no systematic exposition of this matter. The subject report establishes a rigorous theoretical basis for using experimental results to derive the desired characteristics. (auth)

13930 KAPL-1466

Knolls Atomic Power Lab., Schenectady, N. Y.

INCREASING THE EFFECTIVE VOID SPACE IN A

DISPERSION-TYPE NUCLEAR FUEL BY PLASTIC DEFORMATION IN TENSION. S. M. Bishop. Jan. 11, 1956. Decl. Mar. 28, 1957. 20p. Contract W-31-109-Eng-52. $0.35(OTS).

In a dispersion-type nuclear fuel or poison system, the overall dimensional stability of the fuel or poison element under irradiation may be improved by providing void space for the expansion or distortion of the dispersed particles. It is shown that tensile plastic strain increases the void space around particles of UO_2 imbedded in stainless steel; and that, to a first approximation, the amount of void space is proportional to the amount of plastic strain. It is proposed that a controlled, uniform amount of void space could be produced by tensile elongation prior to the final anneal and assembly into the reactor. (auth)

13931 KAPL-M-DWJ-5
Knolls Atomic Power Lab., Schenectady, N. Y.
AIRBORNE ACTIVITY FROM THE VOID SPACE BETWEEN THE SAR REACTOR PRESSURE VESSEL AND INNER WALL OF THE PRIMARY SHIELD TANK. David W. Johnson. [Mar. 1956]. 31p. Contract [W-31-109-eng-52]. $6.30(ph OTS); $3.00(mf OTS).

Air activation in the void space between the SAR pressure vessel and the inner wall of the neutron shield tank has been analyzed. No health hazard results during the normal operation of the reactor from this source. (auth)

13932 KAPL-M-EDL-103
Knolls Atomic Power Lab., Schenectady, N. Y.
DEVELOPMENT OF A FLOW METERING ROD FOR MARK B FLOW MODEL. D. J. Oakley and J. H. Whitman. Sept. 26, 1955. Decl. Mar. 11, 1957. 54p. Contract W-31-109-Eng-52. $9.30(ph OTS); $3.60(mf OTS).

A flow metering rod was developed to simulate the fuel rod in the reactor flow model. In the course of the investigation, it was necessary to reproduce the irregular velocity distribution which occurs at the orifice washer of a fuel rod. The dimensions of the current design of orifice washer proved extremely critical. It was discovered that the flow rate was affected by as much as 15% by the sensitivity of orifice washers to extremely small variations of edge conditions. The resultant design of a flow metering rod which is hydraulically identical to the fuel rod is presented herein. (auth)

13933 KAPL-M-EDL-107(Pt. 2)
Knolls Atomic Power Lab., Schenectady, N. Y.
HOT SODIUM TEST OF THE (S2G) ROTATING PLUGS. Nov. 28, 1955. 281p. Contract [W-31-109-Eng-52]. $45.00(ph OTS); $11.10(mf OTS).

13934 KAPL-M-ELW-3
Knolls Atomic Power Lab., Schenectady, N. Y.
r-θ MULTIGROUP ANALYSIS OF THE PPA-19. E. L. Wachspress. Apr. 9, 1954. Decl. Mar. 13, 1957. 70p. Contract W-31-109-Eng-52. $12.30(ph OTS); $4.50(mf OTS).

The r-θ calculations of criticality, power, fluxes, and sodium activation gave remarkable agreement with quantities measured on the PPA-19. A comparison of measured and calculated powers near the center of the reactor indicated more of a discrepancy here than elsewhere. Calculational techniques were developed to eliminate this discrepancy. (auth)

13935 KAPL-M-ELW-5
Knolls Atomic Power Lab., Schenectady, N. Y.
NUCLEAR CHARACTERISTICS OF A THREE-CYLINDER MOVING FUEL REACTOR. E. L. Wachspress. Dec. 29, 1954. Decl. Mar. 13, 1957. 45p. Contract W-31-109-Eng-52. $7.80(ph OTS); $3.30(mf OTS).

Results of a nuclear survey of three cylinder moving-fuel reactors are presented. Fuel loading, temperature coefficients, xenon effects, and other nuclear parameters are discussed. Power and flux distributions are given for various configurations. (auth)

13936 KAPL-M-GH-2
Knolls Atomic Power Lab., Schenectady, N. Y.
GRAVITATIONAL STRESSES IN THE TOP PLATE OF THE MATRIX BUNDLE. G. Horvay. Oct. 13, 1950. Decl. Mar. 13, 1957. 26p. Contract [W-31-109-Eng-52]. $4.80 (ph OTS); $2.70(mf OTS).

For the case of gravitational loading along the axis of a spoke it is found that the maximum web bending stress is about 1000 psi and the maximum compressive stress is about 360 psi. These stresses are developed in the two webs which face the spoke. The calculations show that for the assumed conditions almost the entire load is carried by the hex column immediately above the spoke, and little stress relief is provided by the adjoining column of hexes. The fact that the stresses are so small must be attributed to the symmetrical loading condition which insures that the bending moment goes thru zero at the weakest cross-section of the most heavily loaded web. There is therefore the possibility that asymmetrical loadings of smaller magnitude or variations in web thicknesses due to manufacturing tolerances will create greater stresses. (auth)

13937 KAPL-M-HH-9
Knolls Atomic Power Lab., Schenectady, N. Y.
THE RELATIONSHIP BETWEEN THE VARIATIONAL METHOD AND ASSOCIATED APPROXIMATE METHODS FOR NEUTRON DIFFUSION IN SLAB LATTICES. Henry Hurwitz, Jr. Oct. 2, 1956. 13p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

The physical significance of the variational method result with constant trial functions is discussed for neutron diffusion in a general array of slabs. In particular the relationship between this method and the three-term linear system which has been used at KAPL for calculating thermal utilization is analyzed. (auth)

13938 KAPL-M-IB-13
Knolls Atomic Power Lab., Schenectady, N. Y.
PPA-13 RIGOR MORTIS. I. Bornstein. Oct. 9, 1953. Decl. Mar. 14, 1957. 5p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

PPA-13 (Preliminary Pile Assembly) is a critical assembly, or mock-up, of a reflector-covered intermediate reactor. Some errors were detected in the multigroup models that had been used to calculate the critical mass. Subsequently, the models were revised, the errors eliminated, and the critical mass recalculated. The recalculated critical mass is 32 kg, as compared to an experimental critical mass of 32.5 kg. The composition of the assembly is tabulated. (L.T.W.)

13939 KAPL-M-JKD-10
Knolls Atomic Power Lab., Schenectady, N. Y.
A MOLTEN-SALT THORIUM CONVERTER FOR POWER PRODUCTION. J. K. Davidson and W. L. Robb. Oct. 15, 1956. Decl. Feb. 26, 1957. 28p. Contract W-3[1]-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

A reactor system incorporating the $Th-U^{232}$ conversion cycle in a homogeneous molten-fluoride reactor is described. The system will operate with an 85% average conversion ratio, for 3,500,000 MWD/ton without fuel purification or fission product separation steps. The reactor core consists of molten Li^7F, BeF_2, UF_4, and ThF_4, and has a critical radius of 175.5 cm or 5.77 feet. The initial critical mass is 398 kg of 93% U^{235}. Additions of ThF_4 and UF_4 (either U^{233}, enriched U^{235}, or Pu^{239}) must be made periodically. Fission product buildup was calculated on the basis

that fission product atoms could not absorb more than one neutron. Based on this optimistic assumption, and utilizing no U recovery, the reactor could be operated for 6.1 burnups with 4 fissions per atom of U^{233} or U^{235} charged initial critical mass plus U^{233} additions. If the uranium is recovered at the end of 4.15 burnups, 7.55 fissions are achieved for each atom of U^{233} or U^{235} consumed. The proposed system would produce electrical power with a fissionable material cost which is approximately 20% of the fuel cost for a reactor which simply burns U^{235}. In addition no processing of the core charge would be required during the design life of the power plant. (auth)

13940 KAPL-M-JS-1
Knolls Atomic Power Lab. Schenectady, N. Y.
EQUIVALENT CROSS SECTIONS FOR PSYCHE. J. C. Stewart. July 5, 1956. 12p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

Fictitious cross sections are derived which permit the replacement of a reactor cell containing a control rod by an equivalent homogeneous cell for purposes of multigroup diffusion theory calculations. Graphs are given for the case of a cylindrical cell which approximates the hexagonal PSYCHE cell. (auth)

13941 KAPL-M-JWW-1
Knolls Atomic Power Lab., Schenectady, N. Y.
AN INTERIM METHOD FOR RAPID ESTIMATION OF FUEL ENRICHMENT FOR VARIOUS REACTOR SIZES. J. W. Weil. Apr 16, 1954. Decl. Feb. 23, 1957. 8p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

13942 KAPL-M-KAK-6
Knolls Atomic Power Lab., Schenectady, N. Y.
PRELIMINARY STUDY OF FAST REACTOR SLUG. K. A. Kesselring. May 10, 1949. Decl. Apr. 8, 1957. 18p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

The results of a preliminary study on a fast spectrum reactor are summarized. Discussions are included on the core composition, the surface area exposed to the primary flowing coolant, properties of the coolant annuli and flow, and the total fuel content. Heat transfer and coolant flow calculations for the fast reactor pin are included in an appendix. (B.J.H.)

13943 KAPL-M-LFC-2(Del.)
Knolls Atomic Power Lab., Schenectady, N. Y.
DESIGN OF FUEL PINS AT ELEVATED TEMPERATURES AND FLUXES. L. F. Coffin, Jr. Oct. 1, 1951. Decl. Mar. 2, 1957. 14p. Contract W-31-109-Eng-52. $3.30 (ph OTS); $2.40(mf OTS).

The power levels of Pu breeder reactors using pin-type fuel elements may be limited by thermal stresses in the pins. The nature of stresses in the pins is considered, and a rational criterion for pin design based on limiting radial deformation is discussed. Laboratory and in-pile experiments for obtaining information on pin design are suggested. (D.E.B.)

13944 KAPL-M-LGB-10
Knolls Atomic Power Lab., Schenectady, N. Y.
THE EFFECT OF REACTIVITY MODULATION AMPLITUDE ON MEASUREMENT OF THE REACTOR TRANSFER FUNCTION. L. G. Barrett. June 29, 1956. 15p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

In the derivation of the reactor transfer function, small signal theory is usually applied and an approximation is made in which the amplitude of the oscillating power is considered small with respect to the steady-state power. In comparing experiment with theory the errors in magnitude and phase may be significant enough to lead to false conclusions. This is especially true in the measurement of power coefficients since the transfer function must be measured at different power levels and vector differences taken to determine the power coefficient. The magnitudes of the expected errors are derived. (G.Y.)

13945 KAPL-M-RE-7
Knolls Atomic Power Lab., Schenectady, N. Y.
MULTI-GROUP CALCULATIONS ON THE FAST LOADING OF THE VARIABLE SPECTRUM REACTOR. R. Ehrlich. Mar. 17, 1950. Decl. Apr. 8, 1957. 38p. Contract W-31-109-Eng-52. $6.30(ph OTS); $3.00(mf OTS).

Multi-group calculations have been made for a spherical model of the variable spectrum reactor with a fast loading with the safety and shim rods at positions both of maximum and minimum reactivity. The results of these two calculations are presented together with a description of the drawing board reactor and the transition to the spherical model. Since this reactor is the current conception of the WMA reactor, the calculations have been labelled WMA-1 for the calculation with all safety and shim rods at the position of maximum reactivity and WMA-1-CE for the control effect calculation (all safety and shim rods dropped the full 22 inches). Section II gives the composition of the actual reactor as obtained from the engineering drawings. Section III gives the composition of the spherical model used for the multi-group calculations. Section IV gives the cross sections used for the calculations. Finally, Section V presents the results and conclusions of the multi-group calculations. (auth)

13946 KAPL-M-RBM-3
Knolls Atomic Power Lab., Schenectady, N. Y.
SCRAM MOTION OF A SPRING DRIVEN CONTROL ROD. Robert B. McCalley, Jr., and Mary C. Ray. Aug. 1956. 30p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70 (mf OTS).

A numerical procedure is presented for integrating the differential equation of motion of a spring driven control rod impeded by Coulomb friction, viscous drag, and turbulent drag. Coding for solution by the IBM 650 Magnetic Drum Calculator is explained and two numerical examples are given. (auth)

13947 KAPL-M-RWL-4
Knolls Atomic Power Lab., Schenectady, N. Y.
RESULTS OF THE 360°, FULL SCALE, WATER FLOW TEST OF THE TOP OF THE KAPL REACTOR WITH HOLD-DOWN PLATE NO. 2. R. W. Lockhart and R. G. Kennison. Sept. 15, 1949. Decl. May 9, 1957. 10p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

A series of 10 photographs illustrates the results obtained from a 360°, full scale, water flow test on the top of the KAPL reactor. The first photograph shows a condition of no flow, while the other pictures cover the range of expected standby coolant flows. Recent design changes incorporated in the test setup are described. Further design changes are suggested on the basis of the test results presented. (C.H.)

13948 KAPL-M-SMS-13
Knolls Atomic Power Lab., Schenectady, N. Y.
A STEADY STATE DYNAMIC ANALYSIS OF THE S3G/S4G NATURAL CIRCULATION STEAM GENERATOR SYSTEM. J. C. Westmoreland. Jan. 9, 1956. 32p. Contract W-31-109-eng-52. $6.30(mf OTS); $3.00(mf OTS).

An analysis of the S3G/S4G natural circulation steam generator has been made to facilitate the prediction of certain dynamic characteristics of the system in terms of its pertinent geometric parameters. A simple two-phase

annular flow model is proposed to permit analytical solutions for the thermal driving head and friction pressure-loss in the riser. This model has been reduced to data for which existing two-phase theories are based and a favorable comparison is presented. The equations from the analysis have been programed for solution by the IBM 650 digital computer and two steam generator configuration have been studied with regard to geometric parameters and also thermodynamic operating state. (auth)

13949 KAPL-M-WGG-2
Knolls Atomic Power Lab., Schenectady, N. Y.
THE COMPOSITIONS OF SAPL IV AND SAPL V. W. G. Gibbard. Feb. 16, 1950. Decl. Mar. 7, 1957. 6p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

Specifications of KAPL Intermediate Power Breeder Critical Experiments are given. Included are the amounts of U^{235}, U^{238}, Be, Na, Fe, and Al used, the dimensions, and the excess reactivity. (M.H.R.)

13950 KAPL-M-WRP-2
Knolls Atomic Power Lab., Schenectady, N. Y.
FUEL PLACEMENT IN S3G ELEMENTS. W. R. Plant. Sept. 27, 1956. 6p. Contract [W-31-109-Eng-52]. $1.80 (ph OTS); $1.80(mf OTS).

Experience has shown that fuel placement in S3G elements is extremely difficult to ascertain from radiographs, notwithstanding the attendant expense in materials and manpower for interpretation. This memorandum summarizes attempts in the non-destructive area to dispense with radiography for determination of this attribute. (auth)

13951 KLX-10000
Vitro Labs., West Orange, N. J.
HOMOGENEOUS REACTOR FUEL REPROCESSING QUARTERLY PROGRESS REPORT [FOR] JANUARY 1–MARCH 31, 1955. JOB 1087. John D. McAdoo, Robert J. Abelson, and Gloria K. Koenig. May 9, 1955. Decl. Feb. 28, 1957. 13p. [For Oak Ridge National Lab.] Contract W-7405-Eng-26, Subcontract No. 535. $3.30(ph OTS); $2.40(mf OTS).

Batch experiments were continued in the investigation of a process for removing iodine from simulated core solution by metallic silver in contact with equilibrium vapor at elevated temperature. The concentration of pentavalent I in oxygenated UO_2SO_4 solution decreases rapidly with a half period of 2 min when the amount of molecular I in solution is negligible. Rate of total I removal from solution after the two forms reach equilibrium was found to correspond to a half period of 20 to 30 min when metallic Ag was exposed to the vapor above a 3.3 cm^2 liquid-gas interface. Rare earth precipitation studies were carried out in an externally heated stainless steel circulating system. Rare earth sulfates dissolved in cold UO_2SO_4 solution were fed to the 300°C loop in 5- to 10-fold excess of the saturation concentration. Almost all of the precipitated material was deduced to have been deposited in the walls of the loop, and about 20% remained there after cooling and draining the system. An ion exchanger, zirconium phosphate, adsorbed 35 mg of cesium/g of exchanger from simulated fuel solution at 95°C. The exchanger was found to have little or no capacity for Ni and Mn ions at 95°C but did remove some Cu. (L.M.T.)

13952 LA-1192
Los Alamos Scientific Lab., N. Mex.
NEUTRON ENERGY DISTRIBUTIONS INSIDE THE FAST REACTOR. Norris Nereson. Dec. 15, 1950. Decl. Feb. 7, 1957. 49p. Contract W-7405-eng-36. $0.45(OTS).

Various experimental methods utilizing nuclear plates have been used in the determination of neutron energy distribution inside the Los Alamos Fast Reactor. Detailed results are given of determinations obtained both by collimating the neutrons from the pot and by experiments carried out directly at the center of the pot. (B.J.H.)

13953 LA-1614
Los Alamos Scientific Lab., N. Mex.
LADY GODIVA: AN UNREFLECTED URANIUM-235 CRITICAL ASSEMBLY. R. E. Peterson and G. A. Newby. Sept. 1953. Decl. Apr. 12, 1957. 49p. Contract W-7405-eng-36. $7.80(ph OTS); $3.30(mf OTS).

A spherical, unreflected U^{235} critical assembly has been in operation since August, 1951. A remotely-controlled mechanical system is used to assemble subcritical components of the sphere, and reactivity is adjusted with U^{235} control rods positioned in the sphere. In addition to investigations of the neutron spectrum of the assembly, observation of the changes of reactivity produced by inserting foreign materials into the assembly, and the determination of parameters such as the temperature coefficient of reactivity, studies have been made of the behavior of the assembly at reactivities above prompt critical. (auth)

13954 LA-1653
Los Alamos Scientific Lab., N. Mex.
NEUTRON DETECTOR TRAVERSES IN THE TOPSY AND GODIVA CRITICAL ASSEMBLIES. G. A. Linenberger, L. L. Lowry, J. A. Grundl, and R. N. Olcott. June 1953. Decl. Apr. 1, 1957. 24p. Contract W-7405-eng-36. $4.80(ph OTS); $2.70(mf OTS).

Neutron detector traverses of the untamped U^{235} (Godiva) and the U^{238}- or Ni-tamped U^{235} (Topsy) metal critical assemblies have been obtained by counting γ-activity of U^{235} and U^{238} fission products, fission fragments of Np^{237} in a spiral chamber, and β-activity of Au and S. At a few positions within the assemblies, and for the U^{238} fission spectrum, cross section ratios of a number of pairs of fissionable isotopes were determined by means of a comparison fission chamber. At the center of Godiva, $\sigma_f(25)/\sigma_f(28) = 6.2$, and this ratio for the Topsy U^{235}–U^{238} assembly ranges from 6.8 at the center to 76 at a radius of $8\frac{1}{4}$". Results of radiochemical analyses for fission and other reaction products are listed. (auth)

13955 LA-1708
Los Alamos Scientific Lab., N. Mex.
MATERIAL REPLACEMENT MEASUREMENTS IN TOPSY AND GODIVA ASSEMBLIES. L. B. Engle, G. E. Hansen, H. C. Paxton, J. C. Hoogterp, and D. S. Young. July 1954. Decl. May 9, 1957. 98p. Contract W-7405-eng-36. $15.30 (ph OTS); $5.40(mf OTS).

This report brings together an extensive accumulation of material replacement (danger coefficient) data for the various oralloy critical assemblies at Pajarito. Corresponding values of effective absorption and transport cross sections are derived. In certain favorable cases, inelastic scattering contributions to the effective absorption cross sections are estimated. Special applications of material replacement data include computation of the relationship between oralloy critical mass and concentration of a diluent, and hydrogen isotopic analysis of heavy water and heavy polythene. (auth)

13956 LA-1874
Los Alamos Scientific Lab., N. Mex.
CALCULATIONS OF THE CRITICAL MASS OF UF_6 AS A GASEOUS CORE, WITH REFLECTORS OF D_2O, Be, AND C. George I. Bell. Feb. 1955. Decl. Apr. 1, 1957. 42p. Contract W-7405-eng-36. $0.40(OTS).

A class of critical assemblies were considered in which the core consists of an oralloy bearing gas at more or less standard pressure and temperature and the reflector is an efficient moderator with a small thermal capture cross section. Age and diffusion theory were applied to compute

the probability that a fast neutron is thermalized by the reflector and then captured by the core. It is first assumed that fast or epithermal neutrons do not interact with the core and it is later shown that this assumption should be a good one for many cases. The reflector may be a thick spherical shell. It is shown that U isotope masses of a few kilograms can be critical within thick reflectors of D_2O, Be, C, or combinations thereof. Curves are attached which enable one to estimate critical configurations. (auth)

13957 LWS-22710

California Research and Development Co., Livermore, Calif.

CONCEPTUAL DESIGN OF A MOLTEN FUEL FBR. R. E. Kelly, L. B. Robbins, and J. B. Stichka. July 2, 1953. Decl. Mar. 4, 1957. 33p. (P-F-13). $15.30(ph OTS); $5.40(mf OTS).

Design drawings and an economic evaluation is presented for a power reactor having a molten $Pu^{239}-Ni-U^{238}$ fuel in a pot-type core. The nuclear properties, heat balance, and fuel processing costs for the reactor are considered. Solutions to the many metallurgical problems are assumed. (D.E.B.)

13958 M-4157

Clinton Labs., Oak Ridge, Tenn.

DESIGN STUDY FOR A BERYLLIUM METAL POWER PILE. H. Etherington, A. Amorosi, G. A. Anderson, E. B. Ashcraft, R. Dick, J. Dietrich, K. Kasschau, and H. E. Stevens. Feb. 3, 1947. Decl. Mar. 7, 1957. 73p. Contract W-35-058-eng-71. $13.80(ph OTS); $4.80(mf OTS).

A design study for a high-temperature, experimental power pile with Be moderator and He as the heat transfer medium is given. The design tends toward uniform dispersal of fissionable material throughout the moderator, and emphasizes conversion of Th. The pile which is of cylindrical form with a horizontal axis, contains a permanent skeletal structure of Be to support the fuel elements and conversion material. The structure is maintained at a maximum temperature of 1050°F by He passing through cooling passages in the structure. A study of the pile physics in relation to design dimensions, per cent voids, conversion, and inventory of fissionable material is included, and a generalized correlation of heat transfer and gas flow with the dimensions and internal structure of the pile is made. The dimensions of the pile are determined by a comparison of these two studies. (D.E.B.)

13959 MIT-5003(Del.)

Massachusetts Inst. of Tech., Cambridge.

POWER PLANTS WITH THERMAL REACTORS. An Engineering and Economic Analysis. Sept. 15, 1953. Decl. with deletions Mar. 5, 1957. 541p. Project Dynamo. Contract AT(30-1)-1359. $2.20(OTS).

Studies were made of the technology and economics of central-station nuclear-fueled power plants using thermal reactors. The principal objectives of the investigation were (1) to determine what are the commercial prospects for nuclear-fuel power vis-a-bis hydrocarbon-fuel power, and (2) to identify and make recommendations concerning the most promising directions leading to technologically and economically feasible nuclear power. Specific investigations were undertaken to determine (1) the characteristics of various nuclear fuel systems in regard to fuel utilization; (2) the characteristics of several combinations of fuel, moderator, and coolant for fixed-fuel reactors; (3) comparisons of fixed-fuel reactors with fluid-fuel reactors; and (4) the optimal type of turbogenerator plant to be used in conjunction with a nuclear-fueled steam generating plant. A nuclear power industry on the scale of 10,000 elec Mw was assumed, thus implying that the costs of now unconventional items will have experienced the reductions which normally come with widespread commercial use. Studies of fuel utilization in fixed-fuel reactors were undertaken to determine (1) whether nuclear reactivity loss interferes with the burnup permitted by radiation damage limitations; (2) whether reuse of depleted fuel elements after processing results in lower fuel feed costs without adverse effects on reactivity lifetime; and (3) what are the unique properties of different fuel cycles. The corresponding results are: (1) Nuclear reactivity loss does not prohibit high fuel burnups. (2) Recycling product results in lower feed costs for Th systems, but not for U systems at present Pu processing costs. (3) Th has advantages over U when the system design permits high specific power. Illustrative designs were carried out for complete power plants based on several types of reactor systems. These designs were executed in sufficient detail to permit cost estimation on the basis of a uniform set of unit costs. Four fixed-fuel reactors, and three fluid-fuel reactors are considered. From evaluations of the illustrative designs it was concluded that several of the systems studied can become competitive in power cost with hydrocarbon-fuel plants after reasonably foreseeable extrapolations of present-day technology, provided that they become established on a sufficient scale so as to permit costs to descend to the levels which commercial use in a sizable industry would permit. Of the types studied, the sodium-cooled, graphite-moderated reactor, the aqueous homogeneous reactor, and the uranium-bismuth liquid-metal reactor seem most worthy of serious effort. Thorium appears to be of great promise as a nuclear fuel both in fixed-fuel reactors and in fluid-fuel reactors. It leads to the desirable combination of low power cost and breeding. The studies of turbogenerator plants show that the relatively low-cost heat available from reactor plants dictates, for purposes of minimum power cost, a turbogenerator of different proportions than those of a conventional plant. For equal throttle conditions, the optimal turbogenerator plant of a nuclear power plant has a smaller turbine-exhaust annulus and a smaller condenser, with a concomitantly lower investment cost, higher turbine-exhaust loss, higher condensing pressure, and lower thermal efficiency. (auth)

13960 MonN-108

[Clinton Labs., Oak Ridge, Tenn.]

PRELIMINARY PROCESS DESIGN REPORT — 1000 PROJECT. M. C. Leverett and J. R. Huffman. May 15, 1946. Decl. Feb. 28, 1957. Includes Supplements 1-7 (Drawings). 103p. 7 illus. Contract W-35-058-eng-71. $31.80(ph OTS); $9.30(mf OTS).

Design criteria are presented for a D_2O-moderated, ordinary-water cooled reactor capable of 30,000 kw output. (D.E.B.)

13961 MonN-383

Clinton Labs., Oak Ridge, Tenn.

SUMMARY REPORT ON DESIGN AND DEVELOPMENT OF HIGH TEMPERATURE GAS-COOLED POWER PILE. C. Roger McCullough. Sept. 15, 1947. Decl. Feb. 27, 1957. 313p. Contract W-35-058-eng-71. $2.40(OTS).

A design for an experimental nuclear power plant utilizing a high-temperature gas-cooled power pile as the energy source is described. The plant consists of the pile, a heat exchanger or boiler, a conventional steam turbine generator, and their associated auxiliaries. Helium gas under pressure transfers heat from the pile to the boiler which generates steam for driving the generator. The plant is rated at a normal output of 12,000 kw of heat and an electrical output of 2400 kw. Provision is made for operation up to 20,000 kw of heat (4000 kw of electrical

output) in the event operation of the plant proves this possible. The pile operates on thermal energy neutrons using uranium enriched to 30% U^{235} as fuel. It is a right circular cylinder with its axis vertical and contains beryllium oxide as the moderating material and permanent skeletal structure. Removable fuel elements consisting of mixtures of enriched uranium and moderating material, which may be beryllium metal, beryllium oxide, or graphite, are located in vertical channels running parallel to the axis of the cylinder. Helium gas at a maximum pressure of 10 atmospheres flows through these channels where it is heated to a maximum temperature of 1400°F. The hot gas is then circulated through a boiler where it is cooled and generates steam, thence is returned to the pile at a temperature of 500°F. (auth)

13962 MTA-43(Del.)
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
AN INVESTIGATION OF FISSIONABLE MATERIAL INVENTORY REQUIREMENTS FOR A LARGE-SCALE NUCLEAR POWER INDUSTRY. D. B. Willmer. June 1953. 27p. Contract AT(11-1)-74. $4.80(ph OTS); $2.70(mf OTS).

The results of a study made to determine the amounts of fissionable material which would be required from outside sources to initiate a large scale central station nuclear power industry are presented. This material would be either U^{233} or Pu to start a breeder cycle of U^{235} for an interim converter cycle producing both power and fuel for an ultimate breeder cycle. The study shows the cumulative amount of fissionable material required up to the date when sufficient breeder fuel is produced to expand the industry at a prescribed rate and the only outside fuel is natural U or Th. Parameters varied include the growth rate of the nuclear power industry, fuel inventory per unit power output of the reactor, breeding gain or conversion ratio, plant efficiency and plant operating factor. The minimization of the fuel inventory per unit of heat capacity in the breeder reactors is shown to be the parameters of greatest importance regardless of whether the nuclear power industry is started directly on the breeder cycle or initiated via an interim converter cycle. The parameters of the converter cycle are of secondary importance in reducing borrowed inventory requirements in comparison with those of the ultimate breeder cycle. (auth)

13963 NAA-SR-96
North American Aviation, Inc., Downey, Calif.
SHIELDING THE NEUTRON PRODUCTION REACTOR. M. Neustadt. Feb. 12, 1951. Decl. Mar. 16, 1957. 41p. Contract AT-11-1-GEN-8. $0.35(OTS).

This analysis of the shielding required for the NPR includes a comparison between a shield with fuel rods passing through it and a shield with no fuel rods passing through it; flux of primary and secondary γ's; radiation emerging through the fuel-rod tube assemblies; the radiation level at the discharge face of the reactor; and heating in the iron thermal shield. (G.A.Y.)

13964 NAA-SR-137(Del.)
North American Aviation, Inc., Downey, Calif.
NATURAL URANIUM REACTORS FOR THE PRODUCTION OF PLUTONIUM AND USEFUL POWER. T. Fahrner, C. A. Trilling, and E. F. Weisner. Sept. 26, 1951. Decl. with deletions Feb. 27, 1957. 56p. Contract AT-11-1-GEN-8. $12.30(ph OTS); $4.50(mf OTS).

The general requirements for a plutonium-and-power reactor are stated. Nuclear characteristics and reactor performance are computed for nineteen graphite and D_2O moderated, natural U metal rod lattices utilizing coolants and structural materials which have possibilities for high-temperature operation. A possible arrangement for a liquid-metal cooled reactor is described. Design objectives for the power plant are stated and the problem of fitting a steam power cycle to a typical reactor coolant temperature cycle is considered. Some tentative conclusions on the economics of plutonium-and-power reactors are presented. (C.H.)

13965 NAA-SR-195(Del.)
North American Aviation, Inc., Downey, Calif.
REACTOR PHYSICS CONFERENCE [HELD AT LOS ANGELES ON] OCTOBER 11, 12, AND 13, 1951. E. R. Cohen and R. H. Sehnert, eds. Mar. 10, 1953. Decl. Mar. 4, 1957. 272p. $45.00(ph OTS); $11.10(mf OTS).

Papers and discussions are reported which were presented at a Reactor Physics Conference held at Los Angeles on October 11, 12, and 13, 1951. This conference brought together physicists with a diversity of interests and problems from the specialized research, design, and operation activities of the national, university, and industrial laboratories engaged in AEC reactor physics work at that time. Major topics discussed included: nuclear data, critical experiments, reactor criticality calculations, reactor kinetics, physics of multi-purpose reactors, and exponential experiments. (C.H.)

13966 NAA-SR-219(Del.)
North American Aviation, Inc., Downey, Calif.
URANIUM PRODUCTION REACTOR (UPR) QUARTERLY PROGRESS REPORT. R. L. Stoker and R. A. Laubenstein. Jan. 29, 1953. Decl. with deletions Mar. 27, 1957. 24p. Contract AT-11-1-GEN-8. $4.80(ph OTS); $2.70(mf OTS).

Preliminary design data are presented for a feasible, economical, production scale converter reactor for producing either U^{233} or plutonium. The reactor core utilizes Al-clad plate type assemblies for the fuel and producer materials. For U^{233} the highly enriched fuel and the producer materials are fabricated into separate elements. Slightly enriched U will be used for Pu production. Light water will serve as both coolant and moderator. Many of the design features are adaptations of the Materials Testing Reactor. 8 figures. (C.H.)

13967 NAA-SR-240
North American Aviation, Inc., Downey, Calif.
PREPARATION OF FUEL ELEMENTS FOR THE NAA HOMOGENEOUS GRAPHITE RESEARCH REACTOR. D. J. Zigrang and G. A. Bennett. Apr. 16, 1953. Decl. Apr. 3, 1957. 36p. Contract AT-11-1-GEN-8. $0.35(OTS).

A procedure is described for the production of homogeneously impregnated graphite blocks for use as fuel elements in the proposed NAA Homogeneous Graphite Research Reactor. Impregnation of 21 inch by 4.18 inch by 4.18 inch graphite blocks with a solution of uranyl nitrate dihydrate (UND) in tertiary butyl alcohol (TBA) is followed by quick-freezing in liquid nitrogen and removal of the TBA by sublimation. Further processing converts the UND to a stable uranium oxide. Uniformity of uranium distribution is confirmed by X-radiographs and chemical analysis. A description of the pilot-plant equipment used is included, together with proposed designs for production equipment. The equipment and procedures are such that the production of fuel blocks could easily be accomplished by the chemical engineering laboratory of a university or by an industrial chemical laboratory. The total cost of fabricating the required number of fuel blocks, exclusive of uranium cost, is estimated to be $18,000, which includes $12,000 for equipment. (auth)

13968 NAA-SR-1347(Del.)
North American Aviation, Inc., Downey, Calif.
SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS

REPORT [FOR] JANUARY–MARCH 1955. SECTION A. A. B. Martin, ed. SECTION B. Guy M. Inman, ed. Oct. 1, 1955. Decl. with deletions Feb. 27, 1957. 108p. Contract AT(04-3)-49. $16.80(ph OTS); $5.70(mf OTS).

Performance and design data are given on the SRE. Temperature coefficients of the effective resonance integrals of U^{238} and Th^{232} were measured and for ¾-in-diameter cylinders, values of + 0.9 ± 0.2% per 100°C for U^{238} and 2.0 ± 0.3% per 100°C for Th^{232} are given. Exponential experiments were made on lattices of 4- and 7-rod clusters, and resulting average neutron fluxes are given in graphical and tabular form. Characteristic analytical data from finished α-rolled, β-heat-treated SRE fuel slugs are given. Observations were made on the dimensional stability of thermal-cycled U alloys at high temperatures. Porous U slugs were thermal cycled in both the α and the β phases, and resulting dimensional changes are shown. Results of thermal cycling jacketed and cored slugs are tabulated. The behavior of Kroll-process zirconium in Na systems was observed, and data on the hardness profile are tabulated. Decomposition of MTR-irradiated terphenyls is observed, as was the gas evolution of MTR-irradiated xylene. Nuclear constants for the SRE were re-evaluated using the average density and enrichment of the U fuel now on hand, and the thermal utilization, resonance escape probabilities, and infinite multiplications are shown. SRE reactor design has been modified so as to lock core components and moderator cells into place. SRE core and reflector data for three cases of SRE criticality are given. Flux and adjoint flux for one of the cases are also shown. SRE neutron flux decay after scram is also shown. The status of reactor construction is discussed. The SRE fuel rod fabrication flow chart is shown. The fuel loader vacuum system and a mockup fuel element are also shown. The design and procurement of materials for the moderator and reflector cells are discussed. Results of tests on the sodium pump development test are given. Engineering tests of sodium flow under SRE conditions are discussed. Preliminary results on control-rod ring heat tests are given. Design and development of safety device systems are discussed. Radiation shielding for the loading face and ring shields of the SRE are shown. Density and thickness requirements for ring and loading face shield concrete are given. The coolant gallery shield plug seal is shown. Results of preliminary experiments with NaK bubblers show O concentration in He after flowing through the NaK bubblers. The interlock region of the fuel handling coffin is shown. (B.J.H.)

13969 NAA-SR-1513

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS REPORT [FOR] JULY–SEPTEMBER 1955. SECTION A. A. B. Martin, ed. SECTION B. J. C. Cochran, ed. Mar. 15, 1956. Decl. Apr. 8, 1957. 141p. Contract AT(04-3)-49. $0.80(OTS).

Loading schedules were proposed for both U and Th–U fuels. Calculations of the isotopic composition of the fuels at discharge from the reactor are tabulated. Measurements were made of neutron distributions in exponential assemblies similar to the lattices proposed for sodium graphite power reactors. Material buckling data for graphite lattices are given. Neutron flux distributions are also given for various lattice arrangements. The development of hot cell facilities is summarized. A temperature monitor assembly for SRE fuel elements is shown. Macroetches of SRE α-rolled β-heat treated slugs indicate a columnar type of structure. Final designs are given for a large hollow slug fuel element. Experiments on the gas evolution and distortion of graphite are briefly discussed. The weight gain of Zr in Na systems was studied. The progress of construction on the reactor is shown. Fuel loading equipment is shown, as is a disassembled SRE fuel element model. Sodium–potassium alloys were prepared and used as bonding material with full length experimental fuel rods loaded with natural U slugs. High-temperature tests on the bond are summarized. A special alignment jig used in moderator canning is shown. Engineering tests on Zr cans are discussed. Data are also given on the tensile stress in Zr. Sodium system designs are discussed. Apparatus is shown for a sodium vapor trap. The pressure distributions in the upper and lower plenum of the SRE were studied. Tests to determine the possibility of galling between the core tank and the core tank sliding pad were continued. A test is described for determining the lubricating properties of MoS_2. A temperature profile through the SRE radiation shield is given. Control and safety rod systems are discussed. Shielding problems are discussed, and data are given on the dose rate above the gap as a function of the thickness of the head shield. Shielding requirements for the radioactive liquid waste disposal system were re-evaluated. Data are given on the radiation level from the hold-up tank as a function of concrete thickness and on the radiation level from the sump discharge line as a function of earth thickness. A system for the analysis of ppm quantities of H in He was developed, and data are tabulated on samples analyzed by this method. Reactor operations are summarized. Diagrams are given for the SRE cooling system and steam plant systems. (For preceding period see NAA-SR-1457.) (B.J.H.)

13970 NAA-SR-Memo-178

North American Aviation, Inc., Downey, Calif.

PRELIMINARY REPORT ON THE CONTROL OF THE CONVERTER REACTOR. L. B. Robinson. Jan. 10, 1952. Decl. Mar. 27, 1957. 9p. Contract [AT-11-1-GEN-8]. $1.80(ph OTS); $1.80(mf OTS).

13971 NAA-SR-Memo-204 (Del.)

North American Aviation, Inc., [Downey, Calif.].

PRELIMINARY STUDY OF POWER CONVERTER REACTORS. Ralph Balent and W. C. Cooley. Jan. 29, 1952. Decl. with deletions Feb. 27, 1957. 13p. Contract [AT-11-1-GEN-8]. $3.30(ph OTS); $2.40(mf OTS).

Preliminary results are reported from calculations of criticality of three proposed designs for the power converter reactor. Tentative design data are summarized in tabular form. Preliminary estimates are included of the cost of producing U^{233} with converter reactors. (C.H.)

13972 NAA-SR-Memo-279

North American Aviation, Inc., [Downey, Calif.].

COMPARISON OF FUEL TEMPERATURES IN PuP AND NRX REACTORS. R. D. Elliott. Apr. 3, 1952. Decl. Apr. 9, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

13973 NAA-SR-Memo-315 (Del.)

North American Aviation, Inc., [Downey, Calif.]

"BOX" TYPE THORIUM ASSEMBLIES AND MULTIPLE-SECTION FUEL ASSEMBLIES FOR THE CONVERTER REACTOR. W. C. Cooley. May 8, 1952. Decl. with deletions Feb. 27, 1957. 10p. Contract [AT-11-1-GEN-8]. $3.30 (ph OTS); $2.40(mf OTS).

It is concluded that the use of box-type Th assemblies leads to a more homogeneous distribution of Th in the core than for other feasible designs, leading to more efficient utilization (lower disadvantage factors) for the fuel and Th. The use of two-piece fuel assemblies for the Cr is desirable in order to reduce the required fuel feed rate by about 22.5 per cent, by interchanging the two fuel sections to

obtain a higher average burn-up, and to permit chemical processing of the fuel sections in the CPP at Arco without having to cut the assembly in two pieces after irradiation. Drawings of a proposed two-piece fuel assembly and Th box design are included. (auth)

13974 NAA-SR-Memo-324
North American Aviation, Inc., Downey, Calif.
ADDITIONAL FLUX DISTRIBUTIONS FOR CONVERTER REACTOR. A. Benton. June 2, 1952. Decl. Mar. 28, 1957. 10p. Contract [AT-11-1-GEN-8]. $3.30(ph OTS); $2.40(mf OTS).

Flux distributions within unit cells were calculated for two essentially different types of lattices proposed for the converter reactor. (C.H.)

13975 NAA-SR-Memo-347
[North American Aviation, Inc., Downey, Calif.]
FLATTENED vs UNIFORMLY LOADED CORE. R[alph] Balent. June 11, 1952. Decl. Mar. 28, 1957. 14p. Contract [AT-11-1-gen-8]. $3.30(ph OTS); $2.40(mf OTS).

The problem of a flattened core vs a uniformly loaded core for the converter reactor project has been reexamined. Several flattened core designs were carried through criticality and power calculations in order to obtain the conversion ratio as a function of critical mass for a given production rate. These results were compared with previous work on uniformly loaded reactors of the same assembly design and heat transfer limitations. The comparison is made for two production rates of 10 kg and 20 kg/mo. The results of this study indicate that the flattened reactors yield higher conversion ratios than uniformly loaded reactors at low critical masses, and uniformly loaded reactors yield higher conversion ratios at higher critical masses. For a production rate of 10 kg/mo, the two systems give identical results at a critical mass of approximately 110 kg, and for a production of 20 kg/mo a critical mass of 250 kg will give the same conversion ratio for the two systems. (auth)

13976 NAA-SR-Memo-617
[North American Aviation, Inc., Downey, Calif.]
CRITICALITY HAZARD IN THE CONVERTER REACTOR MOCK-UP. D. Hetrick. Feb. 20, 1953. Decl. Apr. 8, 1957. 9p. Contract [AT-11-1-GEN-8]. $1.80(ph OTS); $1.80(mf OTS).

Calculations have been performed to determine how much Th can be removed from the mockup of the Uranium Production Reactor before criticality is attained. Two cases are considered: one, 12:6, with an initial configuration of 12 fuel assemblies and 6 Th assemblies; and the other, 9:9, with 9 fuel assemblies and 9 Th assemblies. The total mass of fuel is 2.9 kg in each case. Results are presented in the form of curves of reactivity vs. Th content for calculations in which the Th is assumed to be withdrawn uniformly, and of reactivity vs. number of Th assemblies withdrawn. The results indicate that in the 12:6 configuration, no more than one Th assembly may be safely withdrawn, and in the 9:9 case, no more than two Th assemblies may be withdrawn. (B.J.H.)

13977 NAA-SR-Memo-716
North American Aviation, Inc., Downey, Calif.
RESULTS OF TWO-GROUP THREE-REGION CRITICALITY CALCULATIONS DONE BY THE NATIONAL BUREAU OF STANDARDS ON SWAC. Ralph Balent. June 15, 1953. Decl. Mar. 27, 1957. 20p. Contract [AT-11-1-GEN-8]. $4.80(ph OTS); $2.70(mf OTS).

Two-group three-region criticality calculations are presented which were designed for use in determining an optimum blanket-reflector arrangement for the Uranium Production Reactor and also to facilitate certain types of control calculations. (C.H.)

13978 NNSD-4(Del.)
Newport News Shipbuilding and Dry Dock Co., Va.
NUCLEAR SHIP PROPULSION STUDY. Interim Technical Report. Mar. 1, 1954. Decl. with deletions May 8, 1957. 21p. $4.80(ph OTS); $2.70(mf OTS).

It was found that nuclear power plants were heavier than the corresponding oil-fired plants, but lighter than the oil-fired plants plus associated fuel. The nuclear plants can be accommodated in the spaces occupied by the oil-fired plants; a straight replacement of an oil-fired plant by a nuclear plant without some readjustment of the ship arrangement may not in general work to the best advantage for the nuclear plant, however, due to the special shielding requirements. There appears to be no preferred power range for the application of nuclear power plants as compared with oil-fired power plants. (auth)

13979 NNSD-9(Del.)
Newport News Shipbuilding and Dry Dock Co., Va.
NUCLEAR SHIP PROPULSION STUDY. Interim Technical Report. July 1, 1954. Decl. with deletions May 8, 1957. 37p. $6.30(ph OTS); $3.00(mf OTS).

The application of a 102 Mw pressurized water reactor power plant to a C4-S1-A Mariner merchant vessel is evaluated. The ship, reactor, and steam plant are described, methods of waste discharge are considered, and the effects of shield weight and core lifetime on ship operation are discussed. (D.E.B.)

13980 NNSD-15
Newport News Shipbuilding and Dry Dock Co., Va.
NUCLEAR SHIP PROPULSION STUDY. Interim Technical Report [No. 3 covering period July 1, 1954 to Nov. 1, 1954]. J. B. Woodward, III. Nov. 1, 1954. Decl. May 8, 1957. 33p. $0.40(OTS).

This report covers work done under Study Agreement with AEC dated Nov. 4, 1953.

The operation of the reference design nuclear ship propulsion plant was studied in all situations likely to arise in merchant service. Comparisons were made between oil-fired and nuclear plants in steady-steaming performance, normal maneuvering performance, startup and shutdown, refueling procedures, drydocking procedures, primary system maintenance, and engine department crew requirements. The effect of fission product transients on maneuverability and operation under emergency conditions are also discussed. (M.P.G.)

13981 ORNL-731
Oak Ridge National Lab., Tenn.
A REPORT ON THE SAFETY ASPECTS OF THE HOMOGENEOUS REACTOR EXPERIMENT. Aug. 29, 1950. Decl. Mar. 2, 1957. 67p. Contract W-7405-eng-26. $10.80(ph OTS); $3.90(mf OTS).

A detailed analysis of safety and hazard factors pertinent to the Homogeneous Reactor Experiment No. 1 is presented. Descriptions of the reactor and surrounding environs are included; emphasis is placed on possible accidents arising from mechanical failure, reactivity excursions, and possible explosive recombination of radiolytically produced hydrogen and oxygen. (T.W.S.)

13982 ORNL-1478(Del.)
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROJECT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JANUARY 1, 1953. W. E. Thompson, comp. Mar. 3, 1953. Decl. with deletions Feb. 26, 1957. 133p. Contract W-7405-eng-26. $18.30(ph OTS); $6.00(mf OTS).

Homogeneous Reactor Experiment. The status of HRE operation is discussed. Corrosion of the HRE fuel system is briefly discussed. Power and temperature response curves for the HRE simulator are shown. Boiling Reactor

Research. Diagrams of the BRE shield proposal are given, and γ radiation through the shielding is plotted. The piping drawing for the proposed Teapot design is also given. Data are also given on the response of a boiling homogeneous reactor to changes in steam demand. General Homogeneous Reactor Studies. Tentative design data and the preliminary flow sheet of the Intermediate Scale Homogeneous Reactor are given. Proposed ISHR core vessel and heat exchanges designs are also shown. An analysis is made of the gas separator operation. The flow diagram for the ISHR steam system is also given. Some criticality data are summarized. Flow sheets are given for a test loop to test various sample valves with flowing uranyl sulfate solution. Reactor core development is discussed. Proposed slurry pumps are described. A complete summary of data collected on pin-type corrosion specimens are given. Data on the effect of velocity on the corrosion of type 347 stainless steel are also given. Corrosion data of type 347 stainless steel in oxygenated uranyl fluoride are summarized. Impact test results on commercial purity Ti specimens are compiled. The corrosion of stainless steel by UO_2F_2 and UO_2SO_4 under neutron irradiation is studied. Progress in slurry chemistry is summarized. (D.E.B.)

13983 CP-2342
[Chicago. Univ.] Metallurgical Lab.
TEMPERATURES ON THE SURFACE OF A SLUG JACKET. M. Monet. Nov. 1, 1944. Decl. Feb. 14, 1957. 40p. $6.30(ph OTS); $3.00(mf OTS).

Prior to operating a chain-reacting, graphite-moderated, water-cooled uranium pile at Hanford, the corrosive action of water on uranium was studied. It was necessary to enclose the U cylinders in Al jackets. Preliminary data were available which indicated that Al jacket corrosion might be influenced by the water temperature. Therefore, in order to specify safe operating conditions, it was desirable to know the temperature at any position on the Al jacket of a slug in the tube. Because of the experimental difficulties encountered in the measurement of the correct surface temperature, the problem has been attacked primarily from a mathematical standpoint. Mathematical formulae have been developed for most of the proposed Hanford designs. The purpose of this report was to evaluate and summarize the theoretical and experimental information for calculation of Al jacket surface temperatures for the design now installed at Hanford. Also, the summarized results were to be put into a form suitable for use in routine calculations. As a result of this survey, the Al surface temperature for the Hanford tube and slug design may be calculated within 15°C by employing routine methods and certain simple factors and equations contained in this report. The factors have been assembled from mathematical analyses but agree closely with the existing experimental data. The equations are of the usual type used in heat transfer calculations. Until adequate experimental data are available, it is recommended that the factors and formulae presented be employed in the calculation of the jacket surface temperatures. Experimental work is needed to substantiate the formulae and factors obtained. Also, additional work is needed to determine the surface temperatures necessary to cause early failure of the slug jacket by corrosion, erosion, or cavitation. (auth)

13984 ORNL-1780(Del.)
Oak Ridge National Lab., Tenn.
PHYSICS OF THE HOMOGENEOUS REACTOR TEST—STATICS. M. C. Edlund and P. M. Wood. Aug. 27, 1954. Decl. with deletions Feb. 28, 1957. 69p. Contract W-7405-eng-26. $10.80(ph OTS); $3.90(mf OTS).

The particular features of an adequate theory for a small D_2O-moderated homogeneous reactors such as the HRT are developed. The method is used to perform criticality calculations for the HRT configuration, using core solution of D_2O and enriched UO_2SO_4 and reflector systems of D_2O, D_2O-ThO_2, and low-enrichment UO_2SO_4 solutions. Expected neutron flux distributions and adjoint functions were obtained using 2-group diffusion theory, and graphical results of calculations on the temperature coefficient of reactivity are presented. Initial experiments on the power producing characteristics of the HRT will be made using a D_2O reflector, but later studies will include breeding in the blanket region. In this connection, some attention is given to the build up of U^{233}, and Pa^{233}, as well as the heat generation in D_2O, Th, and U blanket due to the absorption of core gammas. (W.L.H.)

13985 ORNL-1813(Del.)
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROJECT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING OCTOBER 31, 1954. H. F. McDuffie and D. C. Kelly, comps. Nov. 10, 1954. Decl. with deletions May 7, 1957. 202p. Contract W-7405-eng-26. $28.80(ph OTS); $8.40(mf OTS).

Homogeneous Reactor Test. Project status of HRT construction is reviewed and the low-pressure fuel and blanket flowsheet is presented, together with plans of the main heat exchanger, entrainment separator and recombiner-separator. Situations leading to undesirable pressure peaks were analyzed, and maximum safety limits for reactivity increase were established. Miscellaneous radiation hazards were also considered. Full-scale core tests were completed with 9 diffuser screens, and satisfactory hydraulic characteristics were noted. Thorium Breeder Reactor. Conceptual studies of the TBR are largely concerned with the effect of fuel and core parameters on cost factors. Criticality calculations on fuel composition and Xe poisoning problems are considered. Heat generation and thermal stresses were studied for a given core structure. Corrosion. An extensive table of data are given which summarizes the most recent results of UO_2SO_4 corrosion effects on several stainless steels, stellites, and Carpenter 20. In the runs, H_2SO_4, O_2, and Cr additives in the UO_2SO_4 were used. Radiation Corrosion. In-pile and out-of-pile corrosion studies were made on Zircaloy-2 in UO_2SO_4 at various temperatures. Other studies are reported on the corrosion of Chromel-Alumel, Fe–Constantan, and Constantan–Cu thermocouple elements by filtered water, Al_2O_3 bearing materials and Stellite–stainless steel couples by UO_2SO_4, and stainless steel by ThO_2 slurries. Blanket System Components. The beneficial effects of O_2 on the corrosion resistance of stainless steel has been confirmed, and an addition of $Th(SO_4)_2$ to the slurry material has further favorable effects. Experiments show that oxalic acid and sodium silicate have favorable effects on the viscosity of the ThO_2 slurry. Slurry Fuel and Blanket Studies. The addition of UO_2 or CuO to the Th slurries enhanced the catalytic recombination of H_2 and O_2 over the slurry alone. Addition of H_2SO_4 increased Pu(IV) solubility in UO_2SO_4. Further work is reported on the design and efficiency of hydroclone separators for HRT fuel and blanket processing. Supporting Chemical Research. The reactor irradiation reduction of chromate solutions was investigated together with the thermal expansion of UO_2SO_4 solutions and the phase diagram of the $NiO-UO_3-SO_3-H_2O$ system. A method of concentrating N^{15} from a closed-cycle $NH_3-(NH_4)_2CO_3$ system using an exchange column was shown to give a promising separation factor. (C.H.)

13986 ORNL-2128
Oak Ridge National Lab., Tenn.
ARMY PACKAGE POWER REACTOR CRITICAL EXPERIMENT. D. V. P. Williams, D. W. Magnuson, M. L. Batch,

W. R. Johnson, J. K. Leslie, and Dixon Callihan. Aug. 21, 1956. Decl. Mar. 1, 1957. Contract W-7405-eng-26. $0.50(OTS).

A zero power mockup of the Army Package Power Reactor was constructed in the Oak Ridge National Laboratory Critical Experiment Facility. The assembly consisted of 45 fuel elements, in a 7 × 7 array with the corners removed, and was contained in a large water tank. The overall core dimensions were 21 × 21 × 22 in. Enriched U metal foils, 2.5 × 0.002 × 22.0 in., each containing 31.1 g of U^{235} were encased in stainless steel sheets to form the fuel plates. Additional plates of stainless steel and of B encased in stainless steel were used to simulate materials of the APPR-1 core. Each type of plate contained essentially the same amount of steel. When the 18 slots of each element were loaded with plates, the metal volume fraction was 0.197 and was kept constant throughout the experimental program. The number of fuel plates per element varied from 7.5 for the clean reactor to 14.5 for a boron-poisoned assembly. The critical mass of the clean reactor was 10.35 ± 0.05 kg of U^{235}. The addition of 255 g of natural boron to the core, distributed among three plates in each element, increased the critical mass to 20.1 kg of U^{235}. Reactivity coefficients of fuel plates, measured in the clean reactor as a function of position, gave an average fuel value of $0.526/g. The specific mass reactivity coefficient, $m\Delta k/k\Delta m$, was determined over a wide range of reactor loadings. The effects of the heterogeneous distributions of the U and B on the critical masses of the reactor were studied by exchanging special fuel elements with assembly elements in several locations and noting the resulting changes in reactivity. These data and the assumption that the specific mass reactivity coefficient is constant for small changes in fuel loading were used to calculate a value of 9.1 kg of U^{235} for the critical mass of an unpoisoned array of APPR-type elements. Measurements were made of the value of the APPR control rods in various locations, both individually and collectively, in terms of the additional mass of U^{235} necessary to maintain criticality. One eccentric control rod fully inserted was worth 0.68 kg of U^{235}, the central control rod 1.70 kg, and five rods fully inserted required an addition of 12.23 kg to the loading. In addition, the calibration of one eccentric rod by incremental periods gave a value of $3.8. Power and neutron flux distribution measurements were made in all major configurations of the critical assembly. Macroscopic neutron flux distributions were measured by exposing bare and cadmium-covered gold foils at various points in the reactor. Intercell flux distributions were obtained using Dy as the neutron detector. The power distributions were found by measuring the fission fragment activity on Al catcher foils placed in contact with the U in special fuel plates. Catcher foils, exposed both bare and cadmium-covered, in the midplane of the clean reactor, indicated that 91% of the fissions were produced by neutrons having energies less than about 0.5 ev. (auth)

13987 TID-5425

Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.

305-M BUILDING TECHNICAL MANUAL. L. W. Fox. June 1953. Decl. with deletions Mar. 4, 1957. 29p. Contract AT(07-2)-1. $4.80(ph OTS); $2.70(mf OTS).

This manual describes the Savannah River Plant Test Pile, which is used to obtain information on the reactivity effects of reactor components prior to their insertion in the production reactors. (auth)

13988 TID-10030

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

FACILITIES FOR IRRADIATION WITHIN THE MTR REACTOR TANK. C. F. Leyse. June 5, 1953. Decl. Mar. 20, 1957. 69p. (deleted version: AECD-3682). $0.45 (OTS).

Experimental access holes in the Materials Testing Reactor tank, lattice, and reflector are described, and values are given for their fluxes, coolant flow rates, and gamma heating. (D.E.B.)

13989 TID-10077

Dow Chemical-Detroit Edison and Associates. Atomic Power Development Project, Detroit.

INFORMATION REPORT TO THE PROJECT COMPANIES OF THE DOW CHEMICAL-DETROIT EDISON AND ASSOCIATES, NUCLEAR POWER DEVELOPMENT PROJECT. Dec. 1, 1953. Decl. Mar. 11, 1957. 164p. $0.70(OTS).

The development and design of an economic power reactor and associated processing plant are presented. (L.M.T.)

13990 WAPD-A1W(RDM)-397

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]

CONTROL ROD ASSEMBLY MISALIGNMENT STUDY. W. A. Kerr and J. J. Lescisin. Nov. 26, 1956. Decl. Mar. 18, 1957. 8p. $1.80(ph OTS); $1.80(mf OTS).

The vertical force due to misalignment of the Carrier Vessel Reactor control rod assembly which would prevent scram was investigated. (T.R.H.)

13991 WAPD-B(C)-242

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]

DESCRIPTION OF OPERATION OF THE RELUCTANCE MOTOR DRIVE — DRAWINGS 61-J-658 AND 39-A-1201. R. C. Cunningham. Jan. 28, 1952. Decl. Mar. 18, 1957. 5p. 4 illus. $6.30(ph OTS); $3.00(mf OTS).

The primary function of this power system is to provide an ultra-low frequency (⅓ to ⅔ cps), three-phase alternating voltage to drive the reluctance motors coupled to control rods. The associated busses and method of motor connection are shown. A detailed explanation of the operation of the scheme is given. (auth)

13992 WAPD-CP-1098

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PWR CHEMISTRY PROGRAM FOR FISSION PRODUCT REMOVAL. Apr. 20, 1955. Decl. Mar. 25, 1957. 16p. Contract AT-11-1-GEN-14. $3.30(ph OTS); $2.40(mf OTS).

13993 WAPD-CTA-EL-2062

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

PWR (EASE) SIMULATOR STUDY. Series IV (September 1955). Z. H. Calhoun. [Feb. 1956]. Decl. Mar. 18, 1957. 104p. $16.80(ph OTS); $5.70(mf OTS).

Data are presented resulting from an investigation of PWR plant performance following various steam flow changes. The studies include investigations of pressurizer simulation, steam dump system, control system, control system with pressure override, rod drop, Xe poisoning, and operation using only the temperature reactivity coefficient for control. (auth)

13994 WAPD-CTA-EL-2112

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

PWR (EASE) SIMULATOR STUDY. SERIES V (DECEMBER 1955). Z. H. Calhoun. Decl. Mar. 18, 1957. 96p. $15.30(ph OTS); $5.40(mf OTS).

Data resulting from loop flows in the PWR plant are presented. (L.M.T.)

13995 WAPD-CTA(EMX)-230

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

TEST RESULTS OF THE EFFECTS OF CRUD ON

MECHANISM COMPONENTS. T. L. Weybrew. Dec. 19, 1956. 7p. Project 66c19. Contract AT-11-1-GEN-14. $1.80(ph OTS); $1.80(mf OTS).

A number of tests have been conducted to determine possible effects of crud on the operation of mechanism components. Ball bearings, roller nuts, and splines, and a complete S1W Core 3 mechanism were tested. (auth)

13996 WAPD-EM-212

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PWR REACTOR START-UP DESCRIPTION. W. P. DiPietro. Aug. 9, 1954. Decl. Mar. 25, 1957. 23p. $4.80(ph OTS); $2.70(mf OTS).

Reactor startup using remote manual-type control for rods and the chemical concentration control pumps is treated. Cold startup and startup after safety shutdown are considered. The results indicate that cold startup can be accomplished in less than three hours with a maximum fresh water pumping capacity, for chemical dilution, of 200 gpm. Startup after safety shutdown can be accomplished within two hours after the difficulty has been corrected, if the shutdown time is less than one hour. Shutdown of longer than one hour duration may require an estimate of the Xe transient that will occur and fresh water pumping to approximate the Xe buildup in order to be able to start up in a few hours. (auth)

13997 WAPD-FE-464

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

FABRICATION OF MECHANICALLY BONDED FUEL ELEMENT ROD FROM ASSEMBLED COMPONENTS BY COLD SWAGING AND DRAWING. Report on Experiments 11a1(1) and 11a1(2). H. J. Snyder. Oct. 4, 1954. Decl. Mar. 19, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

The fabrication, by cold swaging and drawing or cold swaging alone, of Zircaloy-2-clad U–Mo fuel rods for PWR from assembled components is described. (T.R.H.)

13998 WAPD-FE-784(Rev. 2)

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]

FINAL PROCESS SPECIFICATION FOR COMPACTION OF NATURAL UO_2 FUEL COMPONENTS FOR PWR CORE 1 BLANKET FUEL RODS. B. E. Schaner and J. Glatter. [Dec.] 28, 1955. Decl. Mar. 19, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

13999 WAPD-FE-786

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PRELIMINARY TESTS AND FABRICATION OF THE FIRST HT-1 DEFECT ASSEMBLY TEST. T. J. Kisiel. Apr. 28, 1955. Decl. Apr. 17, 1957. 28p. $4.80(ph OTS); $2.70 (mf OTS).

The first HT-1 defect assembly test is to be run in an MTR loop to determine the effect of one failing U–Mo element on adjacent elements in a rod bundle. This test consists of a bundle of U-12 w/o Mo rods in a square array with the center rod in a sensitized and defected condition. The rods are fastened together in a welded tube sheet design, which provides the high rigidity needed. The center rod was heat treated at 500°C for 10 days and defected in the center with a 0.040 in. hole. Failure should begin after 1 to 4 days under test conditions, the estimate being based on preliminary work in 650°F static water and in the heat thru-put loop. A dummy bundle tested in 650°F static water indicates that the failing rod can produce deflection in adjacent rods. Two bundles of enriched fuel rods were made, one for the HT-1 test and a spare stand-by. The enriched rods were end capped by means of a combination butt and fusion seal welding process and corrosion tested in 650°F static water prior to assembly. The fusion welded bundles were not corrosion tested because of the defect in the center rod. (auth)

14000 WAPD-FE-795

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

STATISTICAL EVALUATION OF EXPERIMENTAL FUEL RODS PRODUCED AT BETTIS. J. G. Goodwin. May 5, 1955. Decl. Mar. 20, 1957. 11p. $3.30(ph OTS); $2.40 (mf OTS).

Radiographs were taken of rod sections to obtain fuel diameter measurements of U–12% Mo pressurized water reactor fuel elements. (T.R.H.)

14001 WAPD-FEM-60

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

STATUS REPORT ON CORE 3 HAFNIUM CONTROL RODS. E. D. Baugh and H. B. Hunter. Feb. 11, 1955. Decl. Mar. 19, 1957. 9p. $1.80(ph OTS); $1.80(mf OTS).

The fabrication, chemical analysis, and nondestructive testing of Hf control rods are described. (C.W.H.)

14002 WAPD-IPC-256 and Add.

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]

PROPOSAL FOR A FUEL MATERIAL TEST IN THE CR VI LOOP IN THE NRX REACTOR, EXPERIMENT CR VI-2. P. C. Daly. [Aug. 1956]. Decl. Mar. 19, 1957. Addendum Decl. Mar. 20, 1957. 14p. $3.30(ph OTS); $2.40(mf OTS).

The addendum to this report was originally issued as WAPD-IPC-308.

The test procedure to be used at NRX Reactor on U–Nb and U–Nb–Zr fuel elements with defects in the Zircaloy-2 cladding is described. (T.R.H.)

14003 WAPD-P-513

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]

SHORT TIME AUTOCLAVE TESTS IN THE MTR. Warren F. Witzig. Sept. 8, 1954. Decl. Mar. 6, 1957. 18p. $3.30(ph OTS); $2.40(mf OTS).

Tests involving the use of an autoclave in the MTR have been initiated to obtain information on the following reactions: the corrosion of Zr–U alloys by H_2O at high temperatures and the behavior of an UO_2 fuel element, with a cladding defect, as it is heated in H_2O. (C.W.H.)

14004 WAPD-P-520

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

EXAMINATION OF MTR FACILITIES FOR PWR PROGRESSIVE FAILURE TESTS. Warren F. Witzig. APPENDIX I. NEUTRON FLUX IN AN ABSORBING CYLINDER. George Hopkins. APPENDIX II. TEMPERATURE DISTRIBUTION IN A CYLINDER WITH NON-UNIFORM HEAT GENERATION. R. E. Grimble and A. Weiss. APPENDIX III. PRESSURE TUBE DESIGN. E. S. Lembersky. Sept. 22, 1954. Decl. Mar. 20, 1957. 27p. $4.80(ph OTS); $2.70(mf OTS).

14005 WAPD-PMM-276

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

SUMMARY OF DATA ON IRRADIATION EXPANSION OF CR-WAPD-MET 2A FUEL RODS. M. L. Bleiberg. Nov. 3, 1955. Decl. Mar. 5, 1957. 15p. $3.30(ph OTS); $2.40 (mf OTS).

The changes caused by irradiation found from measurements on the CR-WAPD-Met 2A prototype fuel rods were in elongation and volume and transverse and longitudinal splitting of the cladding. (F.S.)

14006 WAPD-PWR-CP-1866
Westinghouse Electric Corp. [Bettis Plant], Pittsburgh.
FEASIBILITY STUDY OF A DYNAMIC PRESSURIZED WATER IRRADIATION FACILITY IN A VERTICAL HYDRAULIC RABBIT FACILITY AT THE MATERIALS TESTING REACTOR. Walter Schwartz. [Feb. 29, 1956.] Decl. Mar. 21, 1957. 13p. $3.30(ph OTS); $2.40(mf OTS).

This facility was evaluated for its utilization to study the combined effects of irradiation and corrosion on defectively clad fuel elements. The design is such that these effects may be studied during steady-state irradiation and simulated power cycling. (F.S.)

14007 WAPD-PWR-FE-1138
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
CREEP TESTS ON WATER-LOGGED TUBES. L. F. Cochrun. Jan. 13, 1956. Decl. Mar. 18, 1957. 6p. $1.80 (ph OTS); $1.80(mf OTS).

Creep tests were made on water-logged Pressurized Water Reactor fuel rods to determine the possibility of rod rupture in cases of reactor power level transients. Although the amount of water contained in a water-logged rod does not appear sufficient to cause direct bursting, an exceedingly small leak in the rod would serve to prevent such an accident should the situation arise. (T.W.S.)

14008 WAPD-PWR-FE-1232
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
PREPARATION OF FUEL RODS FOR THE CR-IV X-1-f AND X-1-g TESTS AT CHALK RIVER. T. J. Kisiel. Mar. 23, 1956. Decl. Mar. 18, 1957. 17p. $3.30(ph OTS); $2.40(mf OTS).

The x-1-f test consists of 5 enriched fuel rods in both the defected and undefected conditions, made to fuel tolerances of 0.001 to 0.008 in. on the diameter and 0.060 in. on the length axis. After these rods had been made, calculations indicated that these clearances had a good probability of causing fuel rod failure. It was decided to run the x-1-f test in-pile for a short time to obtain basic fuel rod information and to prepare an x-1-g test in the interim for the desired one year test. The x-1-g rods were fabricated to the new reference tolerances of 0.001 to 0.004 in. diam. clearance and 0.136 to 0.196 in. axial clearance. Both the x-1-f and x-1-g tests were made as closely as possible by PWR production methods. Fabrication details, deviations from PWR process specifications, and pertinent fuel rod data are included. (auth)

14009 WAPD-PWR-PMM-1026
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
METHANE IN PWR CORE 1 FUEL RODS. B. Lustman. Jan. 7, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

Methane gas in PWR fuel rods probably resulted from organic contamination of the tubing. Most of the contamination is eliminated during vacuum annealing. The only serious effect of methane on rod performance is the reaction of the filling gas with excess O_2 in the pellets and subsequent local attack of contaminated regions of the cladding. (L.M.T.)

14010 WAPD-RM-60
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
A PROPOSED METHOD OF MEASURING THE THERMAL NEUTRON ENERGY DISTRIBUTION IN THE INTERIOR OF A REACTOR. Milton Danzker. June 14, 1951. Decl. Mar. 12, 1957. 8p. Contract AT-11-1-GEN-14. $1.80 (ph OTS); $1.80(mf OTS).

An experiment is proposed to determine the neutron distribution in energy in a thermal reactor by using differential absorption to obtain a function from which the distribution function may be calculated. (F.S.)

14011 WAPD-RM-207
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
DETERMINATION OF SHUTDOWN WITH THE AID OF A SOURCE. H. L. Garabedian. Oct. 1953. Decl. Mar. 12, 1957. 26p. Contract AT-11-1-GEN-14. $4.80(ph OTS); $2.70(mf OTS).

The two methods presented for calculating the shutdown of a reactor are designated the sub critical flux determination method and the method of control rod oscillation. The methods are applicable to any reactor of the submarine thermal reactor type such as full-scale ZPR1 assemblies, the SPCA, the Mark I STR, or any similar critical assembly. A brief introductory outline and critique of the two methods for calculating shutdown is followed by expositions of the theoretical analyses associated with each method. (C.H.)

14012 WAPD-T-149
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]
FAST FISSION EFFECT IN CLOSE PACKED LATTICES. R. L. Hellens. [195?]. Decl. Mar. 8, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

A discussion is given of the fast fission effect in H_2O-moderated, slightly enriched reactors designed for power production. Calculations are made to determine the water removal cross section for fast neutrons, and results are given in graphical form. (B.J.H.)

14013 WAPD-TH-114
[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh].
LOCAL SURFACE HEAT TRANSFER COEFFICIENTS NEAR SPACERS IN RECTANGULAR CHANNELS. J. R. Parrette and R. E. Grimble. Aug. 23, 1955. Decl. Mar. 7, 1957. 22p. $0.35(OTS).

Naphthalene mass transfer studies have been made on a large-scale air model to determine the local effect of cylindrical spacers on the surface heat transfer coefficients for parallel plates. Two sizes of circular cylinders and two sizes of elliptical cylinders were investigated at Reynolds numbers from 0.8×10^5 to 3.24×10^5 (based on spacer dimensions). Results indicate that the over-all effect of such spacers is to raise the heat transfer coefficient in their vicinity. A dip below the free stream value of h was noted in only one case where the local value was 80% of the free stream value over a small area. (auth)

14014 WCAP-145
Westinghouse Electric Corp. Commercial Atomic Power, Pittsburgh.
P A R INTERNAL RECOMBINATION. Preliminary General Considerations. J. R. McCord. Apr. 9, 1956. Decl. Mar. 15, 1957. 9p. $1.80(ph OTS); $1.80(mf OTS).

Work is presented on the internal recombination of H_2O in homogeneous reactors. Possible applications to the Pennsylvania Advanced Reactor are suggested. (W.L.H.)

14015 WIAP-M-32
Westinghouse Electric Corp. Industrial Atomic Power Section, McKeesport, Penna.
A PRELIMINARY STUDY OF THE HEAT OUTPUT CAPABILITY OF A PELLET REACTOR. A. R. Jones. Feb. 17, 1954. Decl. Mar. 20, 1957. 12p. $0.30(OTS).

The heat transfer capabilities of an 8-ft-diameter, 10-ft-height cylindrical fluidized or pellet core power reactor are analyzed. Plots of heat output as a function of inlet

H_2O temperature, pellet diameter, and pumping power requirements are prepared. The effects of core diameter on saturation temperatures are also considered. The data indicate increased pellet size, core diameter, and plant design power to be most efficient, however, further study is recommended. (This is a supplement to WIAP-M-8, WIAP-M-12, and WIAP-M-21.) (D.E.B.)

14016 Y-881(Del.)
Oak Ridge National Lab., Y-12 Area, Tenn.
A GRAPHITE MODERATED CRITICAL ASSEMBLY — CA-4. E. L. Zimmerman. Dec. 7, 1952. Decl. with deletions Mar. 6, 1957. 72p. Contract W-7405-eng-26. $12.30 (ph OTS); $4.50 (mf OTS).

An essentially bare graphite moderated critical assembly having dimensions of 51.0" × 44.11" and a critical mass of 52.48 kg of U^{235} was constructed at the ORNL Critical Experiments facility. Using a value for the buckling, B^2, of 0.0018628 cm^{-2} and an extrapolation distance of 2 cm, a multigroup calculation for the assembly gave an effective multiplication of 0.9912. The self-shielding factor due to lumping of the fuel in 0.01 in. discs was measured and found to be 0.94 ± 0.04 compared to a calculated value 0.95. Values of 0.15 ev for the mean fission energy and 27.66% for the fraction of fissions in the thermal group were calculated. Core removal type control rods were calibrated by the "rod-drop" method and by the observation of stable reactor periods. The form of the control rod calibrations curves indicate a contribution due to neutron streaming in the void formed by withdrawing the rod as well as the expected cosine squared distribution. A comparison using a flat strip of cadmium as a poison rod, which left essentially no void when removed, showed good agreement with a cosine squared sensitivity curve. The loss in reactivity as a function of the separation of sections of the assembly showed a loss of multiplication of 0.00635 due to a gap 0.3 in. wide. Bare In and Cd covered In foils were exposed in various parts of the assembly to observe the flux distribution both macroscopically and microscopically. Power distributions were observed by means of Al catcher foils in contact with the U. Comparison between fission rates of bare, Cd covered, and Cd–In covered fuel discs gave values of 2.1 and 3.4 compared to a calculated average value was 2.98. Danger coefficients for Na, Fe, Ni, and Mo were calculated using the multigroup neutron spectrum and the known cross section data. (auth)

PARTICLE ACCELERATORS

14017 LRL-91
California Research and Development Co., Livermore, Calif.
MARK 1 MTA LINEAR ACCELERATOR RUN NO. 8. D. H. Birdsall, P. V. Livdahl, and R. J. Vetterlein. Mar. 1954. Decl. Apr. 5, 1957. 42p. Contract AT(11-1)-74. $7.80(ph OTS); $3.30(mf OTS).

The accelerator was operated with a modified drift tube configuration with the addition of a No. 0 drift tube focusing magnet and a new OFHC (oxygen-free high-conductivity Cu) No. 1 drift tube. Peak proton beam currents of as high as 200 ma were attained at an 8% duty cycle and an average current of 100 ma was achieved at a 72% duty cycle. The first continuous wave beam was accelerated in Mark I on May 26, and subsequently a 50 ma c-w beam was obtained for short periods. At 20 ma c-w, sustained operation was possible. The limiting factor in c-w operation was the injector, but it is believed that the difficulties can be corrected. Beam studies during the run indicated a maximum energy of greater than 13.0 Mev. Essentially the same beam profile and beam center were found with the new configuration as were found in previous runs. Measurements made of the neutron flux indicated a total of 10^{13} neutrons/sec from the Al target cone. (auth)

14018 LRL-106
California Research and Development Co., Livermore, Calif.
FISSION AND CAPTURE EVENTS IN 24 INCH × 24 INCH × $10\frac{1}{8}$ INCH DEEP URANIUM SECONDARY TARGETS BOMBARDED WITH 190-MEV DEUTERONS. H. G. Hicks. Mar. 1954. Decl. Apr. 10, 1957. 12p. Contract AT(11-1)-74. $1.80(ph OTS); $1.80(mf OTS).

Fission and capture events were measured in MTA-type target configurations bombarded with 190-Mev deuterons. The yields were measured for "NaK-cooled" and "air-cooled" simulated arrangements. Al was used to simulate the NaK coolant. The coolant reduced the total number of neutrons formed by about 10% in the case of a U primary target and 25% in the case of a Be target. The U^{235} content in natural U has essentially no effect on the neutron production. (auth)

14019 MTA-16(Rev.)
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
PRELIMINARY OPERATION REPORT OF MARK I MTA LINEAR ACCELERATOR. W. W. Salisbury, P. V. Livdahl, and R. T. Barham. Sept. 1953. Decl. Apr. 1, 1957. 65p. Contract AT(11-1)-74. $10.80(ph OTS); $3.90(mf OTS).

The Mark 1MTA was constructed to prove a new region of accelerator engineering parameters. Performance of the resonant cavity, larger than any previously built, is described. Performance of the drift-tube systems and ion sources are discussed. The simultaneous operation at high-beam current and full proton voltage has not yet been accomplished completely free from sparking. A brief test was performed to examine the cavity after elimination of the shortest drift-tube gaps. Improvement of cavity voltage is reported. (W.L.H.)

14020 MTA-30(Del.)
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
MTA PROJECT TARGET AND PROCESS SECTION QUARTERLY PROGRESS REPORT [FOR] SEPTEMBER THROUGH NOVEMBER 1952. Oct. 1953. Decl. with deletions Feb. 21, 1957. 132p. Contract AT(11-1)-74. $21.30(ph OTS); $6.90(mf OTS).

Results of measurements of fission and capture events produced in unmoderated U blocks with 340-Mev protons are compared with previous results for 190-Mev deuterons. Fission and capture events produced in unmoderated Th blocks by 190-Mev deuterons are also given. Excitation functions were determined for the Cu(p,pn), Be^9(p,p2n), and Be^9(d,d2n) reactions. A neutron-performance analysis of H_2O-cooled U-plate MTA target assemblies by means of two-energy-group neutron-diffusion theory is presented. A mathematical analysis of Pu production with a gas-cooled Be-U target is given both for adjacent and separated primaries and secondaries. The migration area of Po–Be neutrons in H_2O was determined. An expression is developed for determining the stripped neutron flux from the back of MTA targets. A prototype assembly for the study of neutron streaming in the MTA target cavity is described, and some results are presented. New techniques to be employed in reducing the background due to spallations and in measuring angular distributions of neutrons in foils and gases are discussed. Angular distributions of neutrons from a block of U bombarded by 190-Mev deuterons are given. Results of Mark I beam studies are summarized

briefly. Target development studies are reported for primaries of Zr-clad U spheres, H_2O-cooled Zr-clad spheres in the secondary, lattice design, handling and storage, and packed-bed and slurry-fuel lattices as target alternatives. Results of a colorimetric determination of heat release of 190-Mev deuterons in a U target are given. Corrosion of several Al alloys and of irradiated Zr alloys in high-temperature H_2O is reported. Work on a modified Purex Process for processing Zr-clad U primary and secondary target plates is described, including testing of various equipment. (L.M.T.)

14021 NAA-SR-166(Del.)
North American Aviation, Inc., Downey, Calif.
PERFORMANCE OF CAVITY TYPE PRODUCTION LATTICES FOR MTA. W. J. Houghton and E. R. Cohen, Apr. 16, 1952. Decl. with deletions Feb. 26, 1957. 95p. Contract AT-11-1-GEN-8. $16.80(ph OTS); $5.70(mf OTS).

That part of an MTA machine which catches the neutrons has come to be called the lattice. This lattice provides quite a different neutron environment than do catching lattices for reactors, the two outstanding differences being that the neutron source (target) is also a strong parasitic absorber and that the beam hole provides a wide, unobstructed leakage path from the center to the outside of the lattice. The studies presented in this report describe methods of estimating the performance of such lattices and then apply them to actual typical arrangements. (auth)

14022 UCRL-1229
California. Univ., Berkeley. Radiation Lab.
GAP SPLITTER RADIOFREQUENCY POWER LOSS. R. D. Robertson. Apr. 9, 1951. Decl. Mar. 12, 1957. 7p. Contract [W-7405-eng-48]. $1.80(ph OTS); $1.80(mf OTS).

Calculations indicative of the power loss for 75 in. radius gap splitters are given. (D.E.B.)

14023 UCRL-1885
California. Univ., Berkeley. Radiation Lab.
FULL SCALE DRIFT TUBE MAGNET REPORT. D. Sewell and H. Parmentier. July 1952. Decl. Mar. 5, 1957. 31p. Contract W-7405-eng-48. $6.30(ph OTS); $3.00(mf OTS).

Measurements were made on the full-scale drift tube magnets to check values predicted from the model magnets measurements. Measurements of magnetic field, axial forces, power, resistance, water pressure, water flow, electrical vs. magnetic polarity, and thermal cycling were made. (auth)

RADIATION ABSORPTION AND SCATTERING

14024 ANL-4594
Argonne National Lab., Lemont, Ill.
PHOTON ABSORPTION COEFFICIENTS OF LIGHT ELEMENTS AND MIXTURES. R. E. Meyerott and S. A. Moszkowski. Mar. 1951. Decl. Apr. 4, 1957. 21p. Contract W-31-109-eng-38. $3.30(ph OTS); $2.40(mf OTS).

The problem considered is to find a mixture of elements which will act as a filter for radiation by absorbing thermal radiation corresponding to a temperature of 1 to 2 kev, but will transmit radiation of higher frequency. For cases of high temperature and low density, the electrons can be treated as independent and only the coulomb interaction between electrons and nuclei need be considered. Thus, for cases of high temperature and low density, the equations for the various absorption and scattering coefficients take a simple form. These equations are discussed, and it is estimated what mixtures have the desired property of being a filter. (auth)

14025 KAPL-M-MSN-5
Knolls Atomic Power Lab., Schenectady, N. Y.
WIGNER-WILKINS SPECTRA FOR HYDROGEN. D. B. McMillan and M. S. Nelkin. June 7, 1956. 11p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Solutions of the Wigner-Wilkins equation for the thermal neutron spectrum in a monatomic hydrogen moderator were obtained in the IBM-650 for several values of 1/v absorption. The problem is on a production basis for the IBM-650, and spectra can be obtained for arbitrary absorption. The necessary input data to obtain a spectrum are described. (auth)

RADIATION EFFECTS

14026 CF-52-8-148(Chap. IV, Pt. II)
Oak Ridge School of Reactor Technology, Tenn.
REACTOR MATERIALS – CHAPTER IV, PART II. G. E. Evans. [1952]. Decl. Feb. 16, 1957. 77p. Contract [W-7405-eng-26]. $12.30(ph OTS); $4.50(mf OTS).

Some of the basic theory of crystal defects is presented, after which radiation damage to metals is discussed. (L.M.T.)

14027 IDO-16345
Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
ALTERNATE GAMMA IRRADIATION FACILITIES AT THE MTR. A. W. Brown, O. J. Elgert, and W. C. Francis. Jan. 20, 1956. 22p. Contract AT(10-1)-205. (PTR-37). $4.80(ph OTS); $2.70(mf OTS).

Two potential γ irradiation sources have been evaluated for comparison with the present Gamma Radiation Facility utilizing spent MTR fuel elements. The two sources studied were 1) reactor cooling water and 2) an in-pile In loop. The energy available from the reactor cooling water is small compared to spent fuel elements and little potential use of this source is anticipated. The In loop appears considerably more promising and indicates further investigation of other materials (e.g., manganese) and its evaluation by means of a design study. (auth)

14028 KAPL-1091
Knolls Atomic Power Lab., Schenectady, N. Y.
THE STABILITY OF BORON CARBIDE AFTER 15 ATOMIC PER CENT BURNUP. Peter Senio and Charles W. Tucker, Jr. Apr. 5, 1954. Decl. Feb. 25, 1957. 11p. Contract W-31-109-Eng-52. $0.25(OTS).

Hot-pressed boron carbide will withstand a burnup of 15 at. % without physical disintegration. Single crystals show some fragmentation at this burnup. Apparently, the superior behavior of the hot-pressed boron carbide is due to the constraints of adjacent grains which hold the material together even though the individual grains may be fractured. The properties of boron carbide which give rise to its unusual stability compared with reactor fuel materials seem to be its high melting point (strong bonding), high cleavage strength, and open crystal structure. These and other properties of boron carbide should be clues in the search for more stable reactor fuels. (auth)

14029 KAPL-M-JPH-33
[Knolls Atomic Power Lab., Schenectady, N. Y.]
MATERIAL FOR RADIATION DAMAGE COMMITTEE, ARGONNE NATIONAL LABORATORY. John P. Howe. Aug. 31, 1949. Decl. Mar. 13, 1957. 17p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

A program for the irradiation testing of Schenectady Reactor materials is outlined. Preliminary experiments on the radiation effects on U in contact with Na are discussed. (C.W.H.)

14030 MonN-442
[Clinton Labs., Oak Ridge, Tenn.]
PROGRESS ON HIGH TEMPERATURE THERMAL CONDUCTIVITY MEASUREMENTS. L. P. Hunter. Dec. 16, 1947. Decl. Feb. 28, 1957. 66p. $13.80(ph OTS); $4.80(mf OTS).

Measurement of the simultaneous effect of irradiation damage and annealing, as reflected in thermal conductivity measurements of U–Be alloy, BeO and graphite, is made. Methods of impregnation are discussed. (F.S.)

14031 ORNL-864(Del.)
Oak Ridge National Lab., Tenn.
PHYSICS DIVISION QUARTERLY PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 20, 1950. Jan. 8, 1951. Decl. with deletions Mar. 1, 1957. 23p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

Hanford irradiations of Be metal show no changes in electrical resistivity or density within the errors of measurements. New handling equipment for Hanford irradiation of fuel slugs has been constructed. Irradiation of Be–U and Zr–U is continuing. An experiment to measure the resonance absorption of neutrons in U^{238} cylinders has been essentially completed. The results are of special importance for the design of reactors with cylindrical fuel elements. In connection with a cloud-chamber study of capture γ rays, some measurements were obtained on the γ rays from neutrons on U^{235}. The spectral distribution data show a heavy preponderance of γ's toward the lowest measureable energy, 1 Mev. A discussion of the control problems connected with the Bulk Shielding Facility is given, and work on the MTR controls is discussed briefly. (G.A.Y.)

RADIOACTIVITY

14032 WAPD-PWR-CP-1723 and Add.
[Westinghouse Electric Corp. Bettis Plant, Pittsburgh.]
EXPECTED AIR AND WATER ACTIVITIES IN THE FUEL STORAGE CANAL. J. M. Wright. Feb. 1956. Decl. Mar. 21, 1957. Addendum. Mar. 8, 1956. Decl. Mar. 14, 1957. 23p. $4.80(ph OTS); $2.70(mf OTS).

The addendum to this report was originally issued as WAPD-PWR-CP-1896.

The air and water activities which may be expected in the fuel storage canal during the transfer and storage of a core containing defective elements are considered. Only I^{131} and Sr^{90} have the possibility of exceeding the $^1/_{10}$ maximum permissible tolerance limits. (F.S.)

SHIELDING

14033 CF-50-9-118
Oak Ridge National Lab., Tenn.
COMPARISON OF GAMMA INTENSITIES IN LID TANK, 100% H_2O AND 99.4% H_2O–.6% BORON. C. E. Clifford. Sept. 22, 1950. Decl. Feb. 20, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

14034 CF-51-11-96
Oak Ridge National Lab., Tenn.
FAST NEUTRON MEASUREMENTS AT THE BULK SHIELDING FACILITY. H. E. Hungerford and R. G. Cochran. Dec. 10, 1951. Decl. Feb. 16, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Results are compared from three sets of measurements of fast neutrons made along the center line in water at various distances from the Bulk Shielding Facility. A fast neutron dosimeter, a U^{238} fission chamber, and a sulfur foil threshold detector were used for the measurements. Results of the measurements are presented graphically. The thresholds of the various instruments are 200 kv for the dosimeter, 1.5 Mev for the U^{238} fission chamber, and 3 Mev for the sulfur threshold detector. The dosimeter was calibrated against PB-168 Po–Be source. The U^{238} fission chamber and the sulfur were calibrated against a Po–Be source, PB-217. Experiments indicate that there is a discrepancy in calibration between these two sources since there is some difference in their spectra. The dosimeter is usually run behind 3 in. of lead for γ-ray shielding. It was found advisable to do the same thing for the U^{238} fission counter. At large distances the U^{238} fission chamber curve changes slope. This change in slope is probably due to photofission in the U^{238}. The photofission threshold is about 5.76 Mev from γ-ray spectral measurements. There is a peak at 6 or 7 Mev in the gamma ray spectrum from the reactor. This may explain the change in slope of the curve for the U^{238} counter. Considering all the uncertainties involved, the agreement of the three curves is fairly good, both in magnitude and slope. (C.H.)

14035 KAPL-M-IHD-1
Knolls Atomic Power Lab., Schenectady, N. Y.
BORON^{10} SHIELDS FOR Pu α EXPERIMENT. I. H. Dearnley. Feb. 18, 1948. Decl. Feb. 26, 1957. 8p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

A suitable criterion was established for selecting the thickness of B^{10} shield for 200, 1000, and 5000 ev cut-offs in the Pu alpha experiments. (F.S.)

14036 ORNL-1133(Pt. 2(Del.))
Oak Ridge National Lab., Tenn.
THE SHIELDING OF MOBILE REACTORS. [PART] II. E. P. Blizard and T. A. Welton. June 30, 1952. Decl. with deletions Mar. 4, 1957. 54p. Contract W-7405-eng-26. $9.30(ph OTS); $3.60(mf OTS).

The methods of applying results of bulk-shielding measurements to the design of reactor shields are outlined. Geometrical transformations for the more common shapes are derived, as are approximate means of calculating leakage of radiation. As an illustration of the method, the ORNL Lid Tank and Bulk Shielding Facility data are transformed to a standard geometry and compared. Two direct calculations of water attenuation are compared with the experimental data. (auth)

SPECTROSCOPY

14037 IDO-16356
Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
A HIGH RESOLUTION SOLLER SLIT COLLIMATOR FOR USE WITH THE MTR NEUTRON CRYSTAL SPECTROMETER. E. G. Joki, J. E. Evans, and J. R. Smith. June 18, 1956. 10p. Contract AT(10-1)-205. (PTR-92). $1.80(ph OTS); $1.80(mf OTS).

The resolution of the Materials Testing Reactor (MTR) crystal spectrometer has been improved to better than 0.1 μsec/meter by use of a Soller slit collimator of novel construction. The value of such collimators in studying low energy neutron resonances has recently been demonstrated by Sailor et al. The MTR collimator was made by stretching 0.001 nickel foils lengthwise. Coatings of polyethylene approximately 1.5 microns thick on the foils surfaces reduce total reflection. Two collimators with 30 and 40 parallel channels 48 × 2.5 × 0.010 in. (angular divergence–0.7 min of arc) have been constructed. Using one collimator at the reactor face and a second one on the

spectrometer arm, rocking curves having full widths at half maximum of 0.5 min have been obtained at 1 ev with the 221 planes of a calcite monochromating crystal. (auth)

URANIUM AND URANIUM COMPOUNDS

14038 A-7.390.25

[Tennessee Eastman Corp., Oak Ridge, Tenn.]

CALCULATION OF CRITICAL CONDITIONS FOR URANYL FLUORIDE SOLUTIONS. Raymond Murray. Feb. 24, 1947. Decl. Feb. 23, 1957. 15p. $0.30(OTS).

Critical conditions for H_2O tamped and untamped UO_2F_2 spheres and cylinders are calculated using U^{235} isotopic purities of 100, 60, and 30%. (D.E.B.)

14039 A-7.390.27

[Tennessee Eastman Corp., Oak Ridge, Tenn.]

CRITICAL CONDITIONS IN PYREX VESSELS. George W. Schmidt and John W. Morfitt. [1947?]. Decl. Feb. 12, 1957. 13p. $3.30(ph OTS); $2.40(mf OTS).

A method is described for estimating the effect of Pyrex shielding upon the critical mass of aqueous solutions of uranium salts. The method finds particular application in estimating the minimum critical masses of compact arrays of Pyrex carboys. Certain elements of conservatism have been made an integral part of the method, which simplify the mathematical treatment and provide an inherent safety factor. The method follows directly from the so-called "untamped" theory outlined below. (auth)

14040 CF-57-1-3

Oak Ridge National Lab., Tenn.

URANIUM-235 ABUNDANCE BY GAMMA SPECTROMETRY. S. A. Reynolds and J. S. Eldridge. Jan. 2, 1957. Decl. Mar. 12, 1957. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Techniques are described for determining U^{235} abundance by measurement of the intensity of 0.18-Mev gamma radiation. One method involves measurement of the ratio of the intensity of 0.18-Mev radiation to that of ~0.1-Mev radiation. The preferred technique consists of chemical separation of uranium, followed by direct counting of 0.18-Mev photons. Application has been made in analysis of uranium samples of abundances in the range of 0.05% to 93%. Accuracy appears to be better than 3% if the abundance is between 0.7% and 100%. Findings in the self-absorption of 0.18-Mev radiation in uranium oxide are presented. (auth)

14041 IDO-14003(Del.)

American Cyanamid Co. Atomic Energy Div., Idaho Falls, Idaho.

ADSORPTION OF URANIUM ON STAINLESS STEEL. J. C. Ciardelli. Oct. 31, 1951. Decl. with deletions Feb. 22, 1957. 15p. Contract [AT-(10-1)-177]. $3.30(ph OTS); $2.40(mf OTS).

Any U^{235} loss due to adsorption on stainless steel is undetectable with current analytical methods and there is no evidence of its loss due to isotopic exchange. The rate of U adsorption on stainless steel is 18×10^{-9} g/cm^2 and 3 flushes of $Al(NO_3)_3$ solution will adequately remove any U from stainless steel process equipment. (F.S.)

14042 KAPL-M-CVL-4(Del.)

Knolls Atomic Power Lab., Schenectady, N. Y.

SOME CRITERIA FOR SAFE STORAGE OF U-235 FUEL. C. V. Larrick. Jan. 19, 1956. Decl. Mar. 14, 1957. 12p. Contract W-31-109-eng-52. $0.30(OTS).

Criteria are derived for safe storage of U^{235} in arrays that approximate a portion of an infinite slab. This includes storage on the four walls of room. Criteria are also given for safe shipping boxes. (auth)

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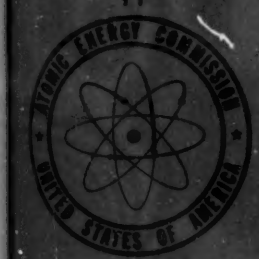

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